Canadian Nuclear Safety Commission · Commission canadienne de sûreté nucléaire

Advisory Committee on Radiological Protection

Radiation Safety Officers Handbook

Part A

INFO-0718

October 2000

Canada

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Advisory Committee on Radiological Protection

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For Information on the indepedent advisory committees contact: Advisory Committees of the Canadian Nuclear Safety Commission, Ottawa, Ontario K1P 5SP.

RADIATION SAFETY OFFICER'S HANDBOOK

PART A

TABLE OF CONTENTS

CHAPTER A1

RADIOACTIVITY A1-1

A1.1 Atoms - the building blocks of matter A1-1
A1.2 The structure of the Atom A1-3
A1.3 Radioactive Transformations A1-6
A1.4 The Nuclear Shell Model and Isomeric Transitions A1-8
A1.5 Alpha Decay A1-10
A1.6 Other Radiations A1-11
A1.7 Decay Schemes A1-11
A1.8 Making Radionuclides A1-12
A1.9 Activity A1-14
A1.10 Exponential Decay A1-14
A1.11 Complex Decays A1-15
A1.12 Series Decay A1-17

CHAPTER A2

RADIATION INTERACTIONS WITH MATTER A2-1

A2.1 Introduction A2-1
A2.2 Charged Particles A2-2
A2.3 Photons A2-4

TABLE OF CONTENTS

A2.4 Attenuation of Photons A2-6
A2.5 Energy Transfer A2-7
A2.6 Sources of Radiation Exposure A2-8

CHAPTER A3
RADIATION DETECTORS A3-1
A3.1 Introduction A3-1
A3.2 Pocket Dosimeter A3-1
A3.3 Gaseous Ion Chambers A3-2
A3.4 Current Ionization Chamber A3-2
A3.5 The Geiger-Müller Counter A3-4
A3.6 The G.M. Counter as an Exposure Meter A3-5
A3.7 Semiconductor Detectors A3-6
A3.8 Scintillation Detectors A3-8
A3.9 Energy Spectroscopy A3-10
A3.10 Thermoluminescent Dosimetry (TLD) A3-11
A3.11 Quantitative Factors A3-13

CHAPTER A4
RADIATION DOSIMETRY UNITS A4-1
A4.1 Introduction A4-1
A4.2 Activity A4-2
A4.3 Exposure A4-2
A4.4 Exposure Rate Constant A4-3
A4.5 Absorbed Dose A4-3
A4.6 Equivalent Dose A4-4
A4.7 Effective Dose A4-4
A4.8 Committed Dose A4-5
A4.9 Annual Limit on Intake (ALI) A4-5
A4.10 Derived Quantities A4-5
A4.11 Problems with Exposure A4-5

CHAPTER A5
CHEMICAL EFFECTS OF RADIATION A5-1

CHAPTER A6
BIOLOGICAL EFFECTS OF RADIATION A6-1
A6.1 Introduction A6-1
A6.2 The Genetic effects of Radiation A6-4

TABLE OF CONTENTS

CHAPTER A7

ADVISORY BODIES A7-1
A7.1 Introduction A7-1
A7.2 The I.C.R.P. A7-1
A7.3 The A.C.R.P., A.C.N.S. and G.M.A. A7-5
A7.4 The I.C.R.U. A7-8
A7.5 The I.A.E.A. A7-9
A7.6 The N.C.R.P. A7-9
A7.7 U.N.S.C.E.A.R. A7-13
A7.8 B.E.I.R. A7-14
A7.9 Associations A7-14

APPENDIX XA1

MATHEMATICAL REVIEW XA1-1
XA1.1 Exponential XA1-1
XA1.2 Logarithms XA1-1
XA1.3 Differentiation XA1-1
XA1.4 Integration XA1-2
XA1.5 Exponential "e" XA1-3
XA1.6 Natural Logarithms XA1-3
XA1.7 Radioactive Decay XA1-4
XA1.8 Photon Attenuation XA1-6
XA1.9 Parent/Daughter Decay XA1-6

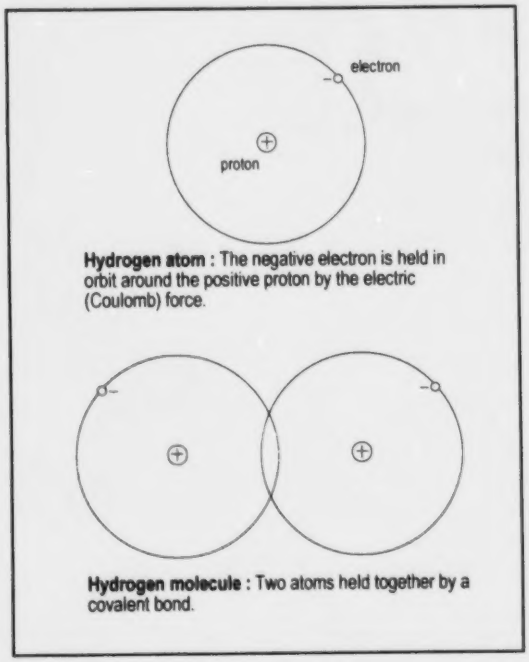
APPENDIX XA2

SAMPLE CALCULATIONS XA2-1
XA2.1 Activity XA2-1
XA2.2 Exposure XA2-3
XA2.3 Shielding XA2-4
XA2.4 Absorbed Dose XA2-5
XA2.5 Effective Dose XA2-6
XA2.6 Committed Dose XA2-7
XA2.7 ALI, DIL XA2-7
XA2.8 Internal Dosimetry XA2-7

CHAPTER A1
RADIOACTIVITY

A1.1 ATOMS - THE BUILDING BLOCKS OF MATTER

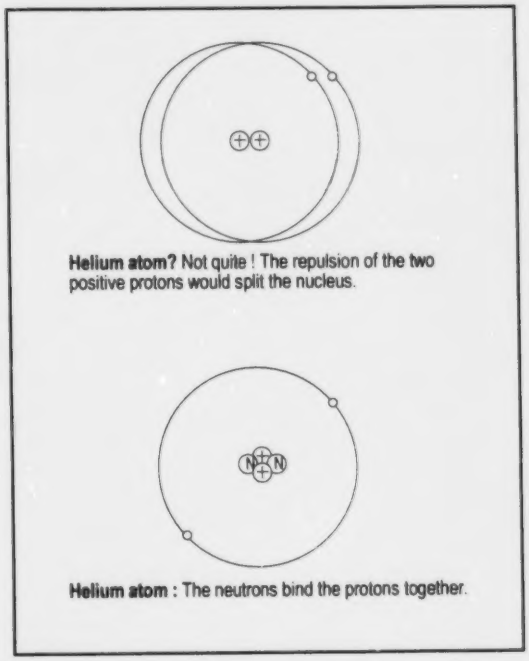
If we push two hydrogen atoms gently together the electrons will form a "covalent bond" and we will have a molecule of hydrogen, usually written chemically as H_2.



Hydrogen atom : The negative electron is held in orbit around the positive proton by the electric (Coulomb) force.

Hydrogen molecule : Two atoms held together by a covalent bond.

To produce a new kind of atom, we have to push the two hydrogen atoms even closer, so that their nuclei fuse together to form a single nucleus, a process called **"nuclear fusion"**. But this nucleus cannot last. The Coulombic force between the two positively charged protons will immediately split them apart, a process known as **"nuclear fission"**. To produce a lasting or stable union between the two protons we require some kind of "glue", sufficiently strong to resist the Coulombic repulsion.



Helium atom? Not quite ! The repulsion of the two positive protons would split the nucleus.

Helium atom : The neutrons bind the protons together.

Enter the neutron, a particle very like the proton, but electrically neutral. Neutrons attract protons and other neutrons via a new type of force or "interaction", called simply **"the Strong Force"**.

If we add two neutrons to our new nucleus, the neutron "pull" will balance the proton "push" and we will have a stable atom of the element helium.

Four Universal forces	
Force	Strength
Gravity	$6x10^{-39}$
Weak nuclear	10^{-5}
Electromagnetic	1/137
Strong nuclear	1

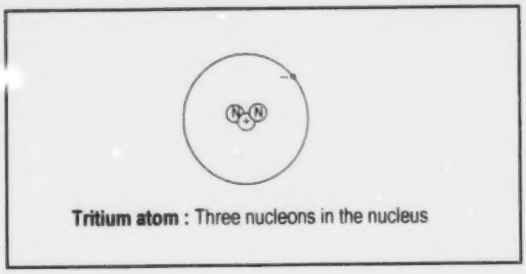
By fusing another hydrogen building block to the helium atom, we can create an atom of lithium, and so on up the periodic table. Each time we add another proton to the nucleus, we must also add an appropriate number of neutrons to balance the extra repulsion.

Neutron / Proton numbers in stable nuclei		
Nuclide	No. Protons	No. Neutrons
helium	2	2
neon	10	10
calcium	20	20
zinc	30	34
tin	40	50
niobium	60	90
mercury	80	122

What would happen if we added too few neutrons, or too many? Then the proton push and the neutron pull would be out of balance, and we would have an unstable nucleus.

A simple example of this is obtained by adding two neutrons to the nucleus of a hydrogen atom. When we specify the number of neutrons and protons in a nucleus, as we have done here, we are defining a particular **"nuclide"**.

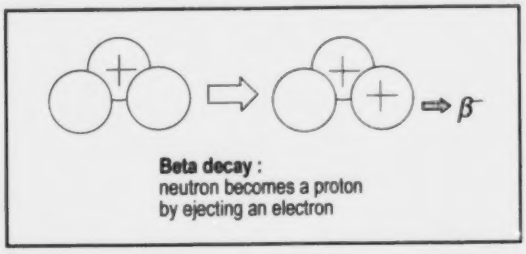
Since this atom still has one proton, and one orbiting electron, chemically speaking it is still hydrogen. To distinguish it from ordinary hydrogen, it is called "hydrogen-3" (abbreviated to H-3 or ^3H) and is said to be an **"isotope of hydrogen"**. This particular isotope of hydrogen has been given the name "tritium", because it has three **"nucleons"** (i.e. protons and neutrons) in its nucleus.



Tritium atom : Three nucleons in the nucleus

The tritium nucleus has too many neutrons, and is therefore unstable. Eventually, one of the excess neutrons will change into a proton. This **"nuclear transformation"** obeys the law of conservation of charge, which says that the total charge before and after a transformation must be the same. Therefore, to change a neutron (0 charge) into a proton (+1), an electron (-1) is also created, and ejected from the nucleus.

When first detected earlier in this century, the fast moving electrons ejected during this process were named **"beta rays"**, and the atoms from whence they came were said (by Marie Curie) to be **"radioactive"**.



Beta decay :
neutron becomes a proton
by ejecting an electron

It appeared, at first, as though pieces of the atom were breaking off, and the term **"disintegration"** was applied to the process, and is still used to denote such a nuclear transformation.

Hydrogen-3, then, is a radioactive isotope (radioisotope) of hydrogen, or, in more general terms, a "radionuclide". It **"decays"** into helium-3, a stable isotope of helium.

Nomenclature

Example : Cobalt-60

Mass no. A (no. of nucleons) → $^{60}_{27}Co$

Atomic no. Z (no. of protons) →

There are mechanisms other than beta decay by which different unstable nuclides find stability, some of which are discussed later in this chapter.

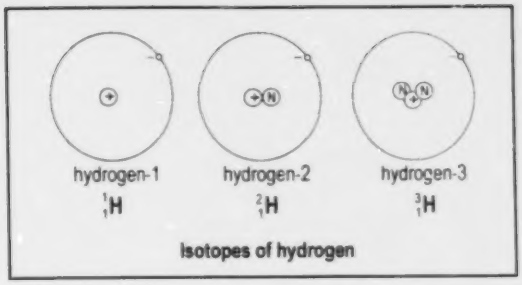
A1.2 THE STRUCTURE OF THE ATOM

There are also naturally occurring radionuclides (about a dozen) which have sufficiently long half lives to have survived since the formation of the earth in measurable amounts. Among these is ^{40}K, the principle source of natural radioactivity in people.

In 1899, J.J. Thompson, having recently "discovered" the electron, suggested that an atom was like a plum pudding, i.e., that it consisted of a ball of positively charged "dough" with sufficient negative electrons mixed throughout to achieve overall neutrality (a chocolate-chip cookie might be a more modern analogy).

However, Thompson's student, Ernest Rutherford, in Montréal shot a beam of alpha particles through a piece of gold foil, and demonstrated that an atom is mostly empty space, with most of its mass concentrated in a small core.

A nuclear model of the atom soon emerged, where the electrons orbit a central nucleus, much as the planets orbit the sun. But it was known from the

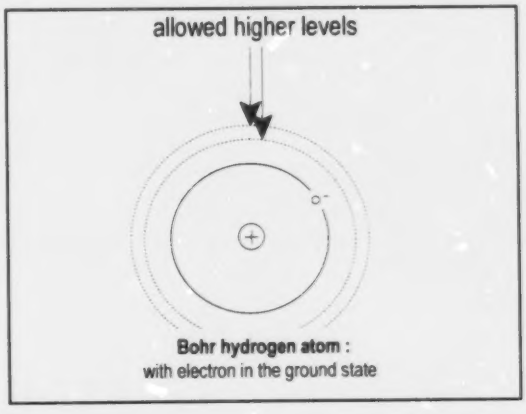


Isotopes of hydrogen

prevailing "classical" theory that electrons radiate energy when they change their direction of motion, and atomic electrons should therefore lose energy, slow down, and quickly spiral into the nucleus.

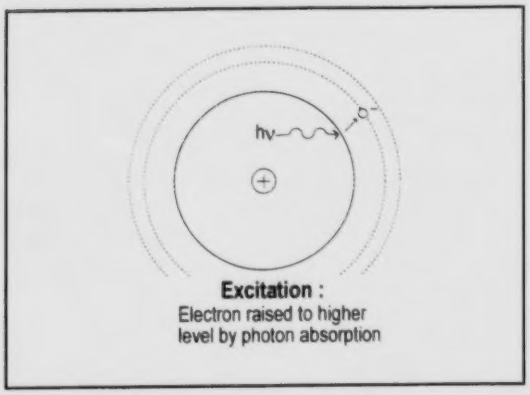
In 1900, Max Planck had suggested that radiation energy is always emitted in small "chunks" called **"quanta"**. In 1913, Niels Bohr applied this **quantum theory** to the hydrogen atom, and obtained theoretical values for the wavelengths of the spectral lines of hydrogen that showed remarkable agreement with experimental observations.

Bohr's picture of a hydrogen atom is shown below. The electron is held in orbit around the nucleus (a single proton) by the electrostatic (Coulomb) force. The electron is confined to certain "allowed" orbits, and normally resides in the orbit closest to the nucleus where it has the lowest possible energy. In this condition, the electron (and the atom) is said to be in its **"ground state"**.

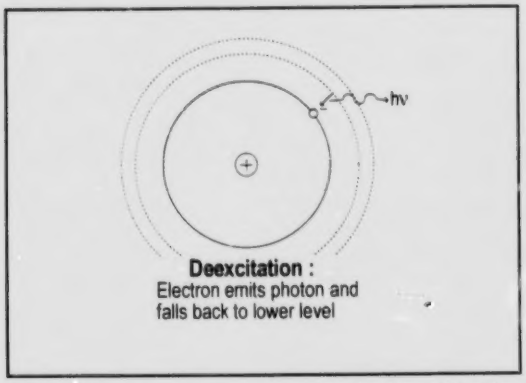


Bohr hydrogen atom :
with electron in the ground state

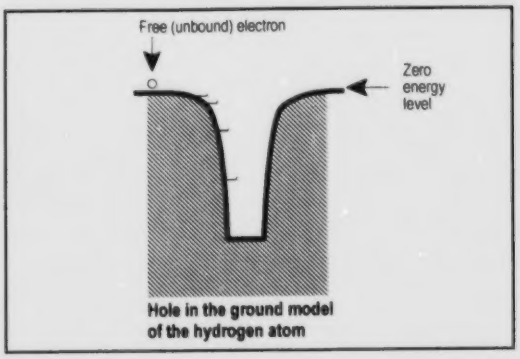
If, and only if, the electron absorbs the exact amount of energy required to lift it to a higher allowed orbit, will it then undergo a transition to this higher orbit, where it is said to be in an **"excited state"**.



The excited electron will seek to fall to the ground state; the exact energy it must lose to achieve a lower orbit is discarded as a single quantum or **"photon"** of electromagnetic radiation. The frequency ν of the radiation emitted is related to the energy E of the photon by the equation E=hν, where h is Planck's Constant (= 4.136×10^{-15} eV.s).

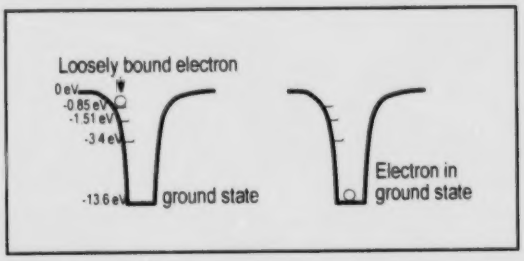


We live out our lives in the gravitational field of the Earth, and consequently have developed an instinctive knowledge of its behaviour. It is sometimes helpful, therefore, to liken an electron moving "up and down" in the electric field of the nucleus to a ball being raised up or falling down in the Earth's gravitational field.



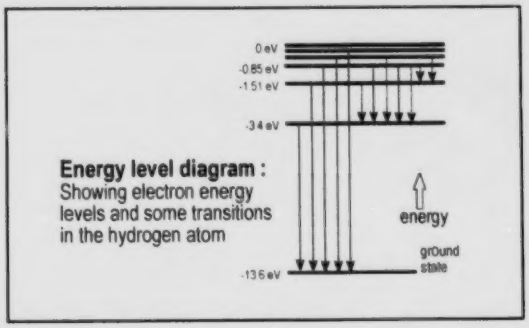
Using this analogy, the allowed orbits of atomic electrons may be thought of as ledges on the side of a hole in the ground. A ball sitting at surface level is not trapped in the hole, corresponding to a "free" electron which is not attached or "bound" to the nucleus. By convention, this level is taken as the zero of energy.

When the ball falls into the hole and lands on the first ledge, its energy is reduced from zero to a small negative value: the electron is now "loosely bound" to the nucleus and a small positive input of energy would suffice to boost it back to the surface.



The ball will fall deeper into the "potential well", moving from ledge to ledge, giving up energy at each step, until it arrives at the bottom of the hole, the ground state.

The **energy level diagram** is a more usual method of depicting this situation. The allowed orbits or energy levels are represented by a series of horizontal lines, with electron transitions shown as arrowed vertical lines.

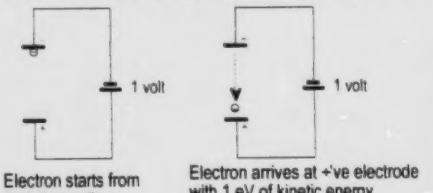


Energy level diagram :
Showing electron energy levels and some transitions in the hydrogen atom

Later in the chapter we shall use similar energy level diagrams to describe transitions in the nucleus of an atom which result in the emission of gamma ray photons.

The electron Volt (eV) unit of energy

One electron volt (eV) is the kinetic energy acquired by an electron accelerated through a potential difference of 1 volt

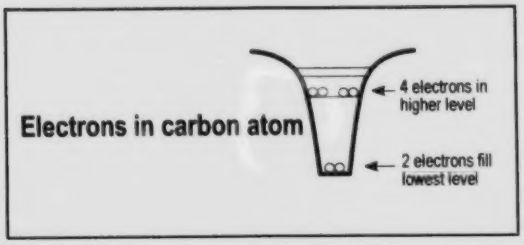


Electron starts from rest at negative electrode

Electron arrives at +'ve electrode with 1 eV of kinetic energy

1 eV = 1.6 x 10^{-19} joules 1 keV = 10^3 eV 1 MeV = 10^6 eV

But first, let us extend our model to atoms having more than one electron, and choose the carbon atom as a relatively simple example. A carbon-12 nucleus is created (in a star) with 6 neutrons and 6 protons.



If and when this nucleus is ejected from the star and slows down (cools), its positive charge will attract and capture a free electron, of which there is always a plentiful supply. This first electron will take up residence in the lowest energy level. The nucleus still has 5 "uncomplemented" protons, and will soon capture a second electron, which will join the first in the lowest state. The lowest state can accommodate only two electrons, and when the nucleus captures a third electron, it is forced to reside in the next level above the lowest.

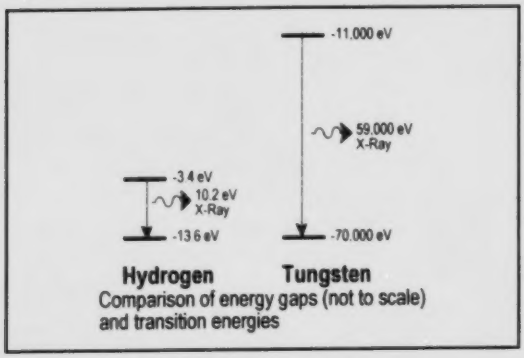
Electron Spin & Pauli's Exclusion Principle

As Bohr's theory was refined to take account of small energy differences due to orbit shape and orientation, and electron spin, the original orbits were replaced by electron "shells". Each shell contains a number of slightly different orbits or energy states which are defined by four "quantum numbers". The total number of electrons a shell can accomodate is governed by Pauli's exclusion principle which states that no two electrons can have exactly the same quantum numbers. For example, the lowest level of an atom can hold two electrons, but their spins must be in opposite directions. Refer to any elementary physics text for more detail.

This process continues until the carbon nucleus has captured its full complement of six electrons to become a carbon atom. Even though some electrons are at higher levels, they cannot fall to lower ones, and the atom is still in its ground state. Such a multi-electron atom, therefore, even when in its ground state, always has electrons poised in higher energy levels, waiting to fall into the vacancy created by the removal of an electron from a lower level.

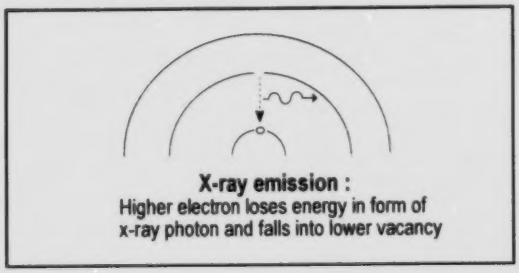
Another important feature of a heavier atom is that its nucleus, by virtue of the larger number of protons, generates a stronger electric field. The potential well (the "hole") is deeper, the inner electrons are more tightly bound to the nucleus, and the energy gaps between the lower levels are greater.

Thus, an electron falling from the first excited state to the lowest state of a hydrogen atom emits a 10.2 eV photon, which is in the ultraviolet part of the

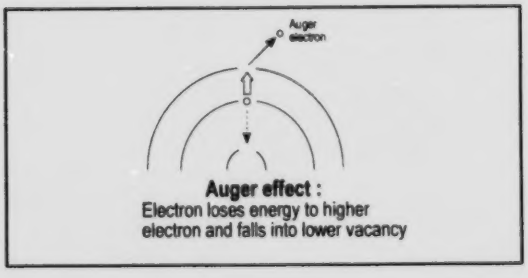


Comparison of energy gaps (not to scale) and transition energies

electromagnetic spectrum, whereas the equivalent transition in a tungsten atom produces a 59,000 eV photon, which is an x-ray.


X-ray emission :
Higher electron loses energy in form of x-ray photon and falls into lower vacancy

An electron seeking to occupy a vacancy in a lower energy state has another way of shedding energy besides emitting a photon: it can transfer the surplus energy to another orbiting electron. This second electron is ejected from the atom taking the surplus energy with it. This process is known as the **Auger effect** (pronounced "Oh-Zhay").



Auger effect :
Electron loses energy to higher electron and falls into lower vacancy

The ejected electron carries the same "characteristic" energy as an x-ray photon produced by the same transition, minus the relatively small amount of energy the electron must expend in climbing out of the potential well away from the positive nucleus.

How does the first electron transfer energy to the second electron? The process cannot be explained in terms of the Bohr model which depicts electrons as particles moving in separate orbits. In models which superseded the Bohr model, electrons are treated as waves. In such a model, the wave functions can overlap, and energy transfer between such overlapping electrons is possible.

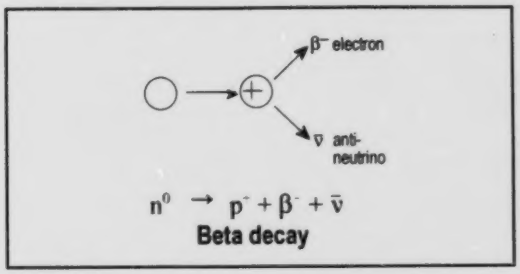
Several radioactive decay processes result in the removal of an "inner shell" electron from the atom, creating a vacancy, and leading to the emission of extra radiation in the form of x-rays or Auger electrons.

A1.3 RADIOACTIVE TRANSFORMATIONS

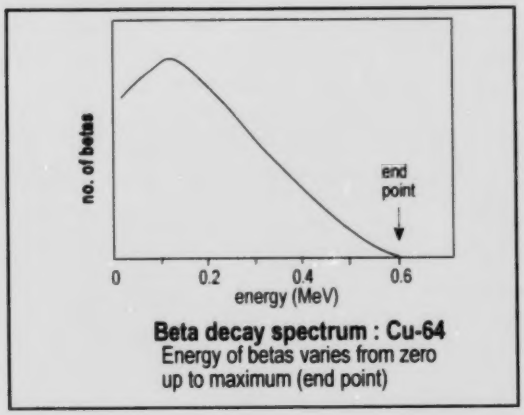
We have already seen that an unstable or radioactive nucleus is the result of an imbalance in the neutron/proton ratio, i.e., too many neutrons or too many protons. We will now examine some of the transformations a nucleus may undergo to correct such an imbalance.

A1.3.1 NEUTRON EXCESS: BETA DECAY

An excess neutron is transformed into a proton and an electron. If we examined the mass of the original nucleus, and compared it to the total mass of the final nucleus plus the newly created electron, we would find that a small amount of mass had disappeared. It is this missing mass, transformed in accordance with Einstein's famous equation, $E = mc^2$, that supplies the energy needed to eject the beta particle.

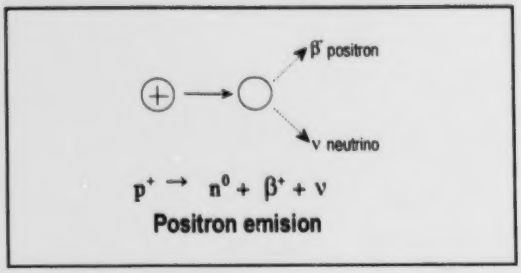


However, experimental measurements show that the beta particle does not receive all of the energy available from the mass conversion. It turns out that the total available energy is randomly divided between the beta particle and another particle called a **"neutrino"**, which is created at the same time. Therefore, beta particles emitted by a particular radionuclide have various energies which range continuously from almost zero up to the maximum transition energy available from the mass conversion.



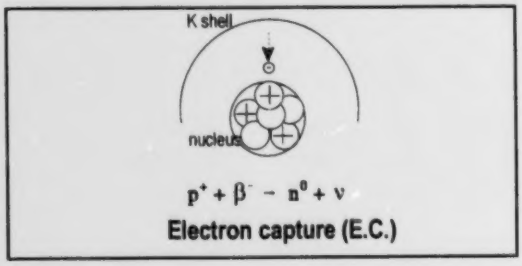
A1.3.2 PROTON EXCESS: POSITRON EMISSION

An excess proton is transformed into a neutron plus a **positron** and a neutrino. A positron is a positive electron, the anti-matter twin of the more familiar negative electron.



A1.3.3 PROTON EXCESS: ELECTRON CAPTURE (E.C.)

As an alternative to positron emission, a nucleus may transform a proton into a neutron by capturing one of its own orbiting electrons. The electron captured is usually from the shell closest to the nucleus (the K shell).

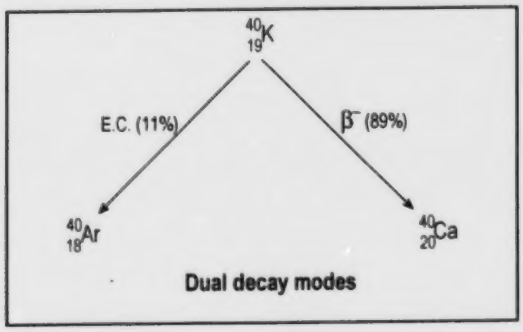


As with the Auger effect, mentioned previously, the simple Bohr model of the atom does not allow an explanation of this process. The more recent wave-mechanical model allows the electron wave to overlap the nucleus wave. Put differently, the electron has a small but finite probability of being found inside the nucleus, which makes its capture seem more reasonable. A neutrino is also created during the process.

E.C. is more common than positron emission, especially in heavier atoms where the K shell electrons are closer to the nucleus. Since the capture process leaves a vacancy in the K shell, which will be filled by a higher electron, E.C. results in x-ray or Auger electron emission.

A1.3.4 SIMULTANEOUS PROTON AND NEUTRON EXCESS

When we previously discussed nuclear stability in terms of the number of protons and neutrons in a nucleus, we failed to mention another, subsidiary effect, which is: nuclei with an even number of protons, or an even number of neutrons, or better yet, even numbers of both, are more stable than nuclei with odd numbers of protons or neutrons.



Dual decay modes

Thus, $^{40}_{19}$ K, for example, which is an "odd-odd" nucleus with 19 protons and 21 neutrons, has two possible ways of becoming a more stable "even-even" nucleus: it can change a neutron into a proton by beta decay to become $^{40}_{20}$Ca, or it can change a proton into a neutron (by electron capture) to become $^{40}_{18}$Ar. For K-40 and several other odd-odd nuclides, this is such a balanced choice that a fixed proportion of the nuclei will undergo one or the other transformation.

A1.4 THE NUCLEAR SHELL MODEL AND ISOMERIC TRANSITIONS

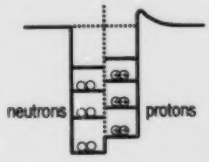
In the Bohr model of the atom, it is easy to visualize electrons moving in empty space around a single, distant nucleus, with little or no interaction between one electron and another. It is more difficult to accept a similar picture for the structure of the nucleus itself. How can protons and neutrons revolve in orbits inside something as dense as a nucleus, without colliding and producing chaos?

Many features of nuclear behaviour can be explained on the basis of such a model, often called the "single particle" or **"shell model"** of the nucleus. Some credibility was added to the idea when Weigner pointed out that, according to the quantum theory, nucleons (i.e. neutrons and protons) can collide without having an effect on each other.

Two differences between the nuclear shell model and the atomic shell model are relevant to our discussion. First, because the nuclear strong force is about 137 times more powerful than the atomic Coulomb (Electromagnetic) force, the nuclear potential well (the hole) is much deeper, and the gaps between nuclear energy levels are much larger.

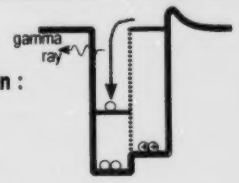
Second, there are two different sets of energy levels in a nucleus, one set for protons and another set for neutrons, because the protons are charged and the neutrons are not.

Hole in ground model :
C-12 nucleus in its ground state. Note different energy levels for neutrons and protons.



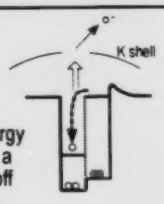
We will return to these two sets of levels in a moment, but first we will discuss nucleon transitions in much the same terms as we previously discussed atomic electron transitions.

When a nucleon falls from a higher to a lower energy level, it emits a photon of electromagnetic radiation, just as an atomic electron does, but the energy of the nuclear photon is much greater, in the range of the electromagnetic spectrum we call **"gamma rays"**. Nuclear energies are usually measured in keV or MeV, whereas atomic (electron) energies are in the eV or keV range.



Gamma ray emission :
Representation of a nucleon falling to a lower energy state by emitting a gamma ray photon.

Instead of emitting a gamma ray, the nucleon may transfer the energy it needs to lose to an inner shell electron (remember those overlapping wave functions?) which is ejected from the atom, taking the excess energy with it. This nuclear analog of the Auger effect is called **Internal Conversion** (I.C.), because it was originally supposed that a gamma ray was first emitted, which then knocked out or "converted" an electron in the same atom via the photoelectric effect (see chapter A2.3). There is now plenty of evidence available to prove that this, in fact, does not happen.



Internal conversion :
Nucleon transfers excess energy to atomic electron and falls to a lower level. Electron carries off the excess energy.

These secondary transitions that proceed either by gamma ray emission or by internal conversion are called **isomeric** transitions. The "parent" and "daughter" nuclei are said to be **isomers**. They are, of course, the same nuclei, having the same numbers of protons and neutrons before and after the isomeric transition. The only difference is that the nucleus is in a higher energy or "excited" state before the transition.

The nucleus may remain in the excited state for some time before undergoing the isomeric transition. Lifetimes for excited states range from about 10^{-16} seconds to almost a million years, depending on the nuclide and the excited state involved. Excited states with long lives, long enough to be measured experimentally, were originally thought to be separate forms of the nuclide, and the term "isomer" was reserved only for these. But with hindsight, such an "isomer" is not a separate species, but only a long lived, or **metastable** energy level. So we can say that Tc-99m, for example, is a metastable level of Tc-99, or, using the older jargon, an isomer of Tc-99.

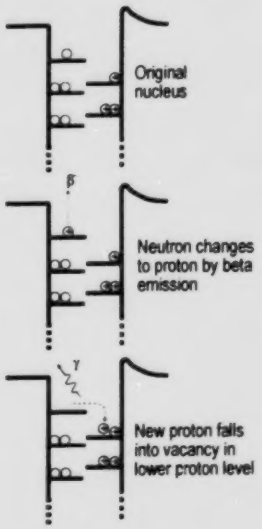
Metastable excited states

Transitions involving photon emission are governed by a set of "selection rules". A nucleon may be "stuck" in an excited state for some time if its transition to a lower level is "forbidden" by these selection rules.

We now turn our attention to the second difference between the nuclear and atomic models, and refer to the "hole in the ground" representation of a carbon-12 nucleus, with one side of the hole reserved for neutrons and the other side reserved for protons. Note that the proton side has a raised "lip". This represents the repulsive electric force, or "Coulomb barrier" that an incoming proton must surmount before it is captured (i.e. falls into the hole) and becomes part of the nucleus. We will mention this again when we discuss the production of artificial radionuclides by proton bombardment.

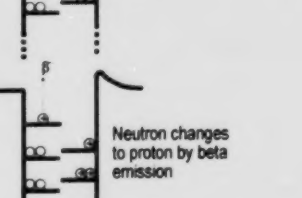
This representation allows us to look at several aspects of nuclear stability from another viewpoint. Nucleons, like atomic electrons, will always seek the lower possible energy level. If we have a nucleus with the lower neutron levels full, but with a vacancy in a lower proton level, then a neutron in a high level may change into a proton so that it can occupy this vacancy in the lower proton level. This is another way of describing beta decay, followed by gamma ray emission as the new proton falls to a lower level.

Note also that as we go to heavier nuclei (more nucleons), there comes a point where there are more neutron levels available at lower energies than equivalent proton levels. Hence, the heavier stable nuclei will have more neutrons than protons.

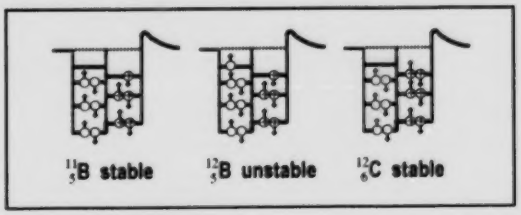


Beta decay :
Depiction of beta decay using energy level concept

Finally, note that Pauli's exclusion principle applies to nucleons. Each sub-level can accommodate two nucleons, provided their spins are in opposite directions, which means that nucleons, like the animals entering the Ark, tend to go in pairs. If a nucleus is synthesized with an odd proton alone in an energy level, and an odd neutron alone in its energy level, one will transform to join the other at a lower level, as shown in the preceding beta decay diagram. This explains why stable nuclides, where possible, tend to have even numbers of nucleons.

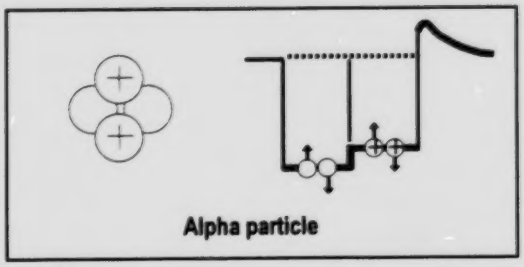


Due to repulsive Coulomb force, proton levels are more widely spaced than neutron levels. At N = Z = 22, proton levels overtake neutron levels. Thereafter, there are more lower neutron levels available than proton levels. Hence, larger nuclei have more neutrons than protons.



A1.5 ALPHA DECAY

A prime example of the even-even effect at work is the **alpha particle**, which consists of two protons and two neutrons, i.e. a helium nucleus. This group is so tightly bound together that it is often energetically more favourable for a nucleus to eject a whole alpha particle than to break it up and eject a single proton or neutron.



This is especially true of very large nuclei. Here, the neutrons are spread far apart, and the short range attractive force holding the nucleus together is less effective. Of course, the protons are also spread apart, but the proton repulsive force operates over a longer distance, and is not so diminished. Such a nucleus needs to reduce its size, and alpha emission is one way of doing this (nuclear fission is another).

Hence, $^{209}_{83}$Bi is the heaviest stable nuclide and all nuclei with Z > 83 and A > 209 spontaneously transform themselves into smaller, more stable nuclei through the emission of one or more alpha particles.

Alpha emission is the primary mechanism by which the three naturally occurring radioactive series arise from their three parent radionuclides. For example, the parent nuclide $^{238}_{92}U$, decays by alpha emission to $^{234}_{90}Th$, which in turn decays by beta emission to $^{234}_{91}Pa$, which decays by beta emission to $^{234}_{92}U$ which decays by alpha emission to $^{230}_{90}Th$, and so on down to the stable nuclide $^{206}_{82}Pb$.

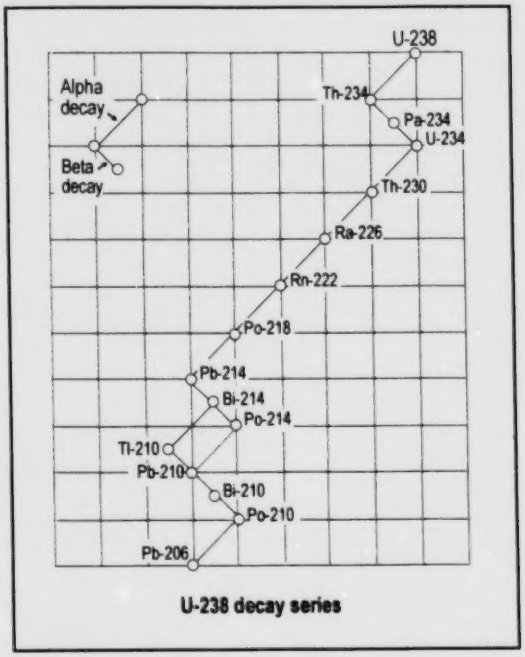
A1.6 Other Radiations

One might reasonably ask: if a nucleus has too many protons, why does it not simply eject one, instead of converting it to a neutron? Likewise, why not eject a surplus neutron, rather than transform it into a proton? The answer to this question lies in a more detailed analysis of the mass / energy change between the parent and the daughter nuclide which would result from such an emission. For our purposes, we can simply say that it is easier for a beta particle, for example, to climb out of the nuclear potential well than for a neutron or proton (betas are not bound by the strong nuclear force, whereas neutrons and protons are). The alpha particle, as already mentioned, is a special case, because of its "binding energy". But even alpha particles would have difficulty leaving the well were it not for their ability to "tunnel" through the wall, a characteristic which can only be explained in terms of the wave mechanical model of the nucleus.

Neutron radiation is produced in a nuclear reactor as a result of nuclear fission. Less bulky sources of neutrons are available by mixing an alpha-emitting radionuclide with beryllium, which absorbs the alphas and emits neutrons according to the nuclear reaction:

$$^{9}Be + {}^{4}He = {}^{12}C + {}^{1}n$$

Almost any other atomic particle, especially protons, can be accelerated in various types of particle accelerator to produce beams of radiation directly, or by letting the beam strike a target, to produce other radiations via various nuclear reactions. Few, if any, of these radiations are pertinent to our present purpose.



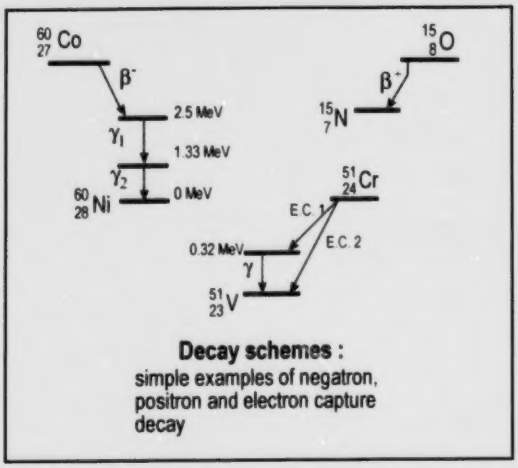
U-238 decay series

A1.7 Decay Schemes

The various processes that a radionuclide undergoes can be summarized in a standard diagram, call a **decay scheme**. We start with a horizontal line at the top of the diagram, representing the initial or parent nuclide in its ground state. The primary nuclear transformation (e.g. a neutron transforming into a proton) is depicted by a downward sloping arrow. If the atomic number (the number of protons) is increased by the transformation, the arrow slopes to the right. If the atomic number is reduced by the transformation (e.g. a proton to neutron change), the arrow slopes to the left.

The arrow ends on one of a number of horizontal lines which represent the nucleon energy levels of the new (daughter) nuclide. If the arrow goes directly to the lowest of these levels, then the daughter nuclide is formed in its ground state, and no isomeric transitions will ensue. If the arrow

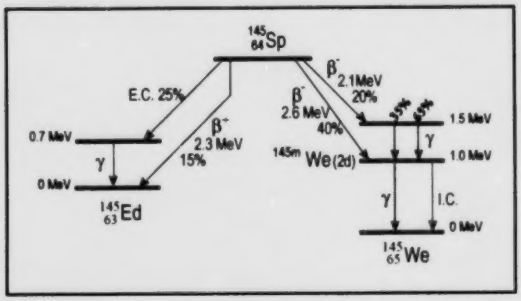
goes to one of the intermediate levels, then isomeric transitions will follow as the nucleus seeks its lowest possible energy configuration.



Decay schemes :
simple examples of negatron, positron and electron capture decay

Radionuclide decay is a random process. In a collection of identical unstable nuclei, some may follow one route to stability, while others follow an alternate route, if such an alternate is available.

To illustrate this, and other features of a decay scheme, consider the decay of 1000 nuclei of a hypothetical nuclide with the fictitious name Sportium-145 (Sp-145).



We might see that 600 of them undergo initial negatron (negative beta) decay to produce the fictitious daughter nuclide Westium-145 (We-145), while the other 400 decay to Edion-145 (Ed-145).

Of the 600 that undergo negatron decay, 200 (i.e. 20% of the original 1000) form the daughter nucleus We-145 in its second excited state, which is 1.5 MeV above the ground state. The other 400 negatron decays (40%) form the daughter in the first excited state, 1.0 MeV above the ground state. The end point energies of the negatrons leading to each excited level are shown to be 2.1 MeV and 2.6 MeV.

So, the 1.5 MeV level of We-145 will be "populated" by 200 nuclei; 70 of these (35% of the 200) fall directly to the ground state, emitting 70 gamma ray photons in the process, each photon having an energy of 1.5 MeV. The remaining 130 nuclei (65% of the 200) fall to the first excited state, emitting 0.5 MeV photons in the process.

The first excited state is a metastable level (We-145m) with a half life of 2 days. 400 nuclei arrive at this level directly from the 400 negatron decays, and are joined by the 130 nuclei mentioned above, for a total of 530. These nucleons eventually reach the ground state by internal conversion with a 2 day half life.

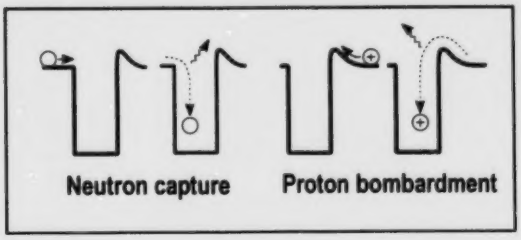
Turning to the other branch of the initial decay, 250 nuclei (25% of the original 1000) decay by electron capture to the first excited state of Ed-145, while the other 150 (15%) go directly to the ground state by positron emission. Note that the positron emission arrow has an initial vertical run, denoting the 1.02 MeV loss of energy that goes into the creation of the positron via Einstein's energy to mass conversion formula.

The decay scheme, therefore, with a little mathematical effort, allows us to determine how much of each kind of radiation will be emitted from a particular amount of any given radionuclide.

A1.8 MAKING RADIONUCLIDES

We have seen that an unstable or radioactive nucleus is the result of having too many neutrons or too many protons. So, to make radionuclides, all we have to do is add extra neutrons or protons to ordinary, stable, atomic nuclei.

We can use our hole-in-the-ground model of the nucleus to examine this process. A neutron, being neutral, has no Coulomb barrier to surmount and can approach a target nucleus at a casual pace and trickle into the well. As it falls, its excess energy is discarded by gamma ray emission. This is called a **neutron capture** reaction or an **n-gamma** reaction. An example of this might be the production of radioactive Mo-99 by adding a neutron to stable Mo-98. This reaction would be written as $^{98}Mo\ (n,\gamma)\ ^{99}Mo$.



For a proton to enter a nucleus, it must be moving fast enough to roll up and over the top of the Coulomb barrier (i.e. overcome the repulsion of the protons already in the nucleus). Once the proton is inside the nucleus (in the hole) the short range strong force keeps it there. A particle accelerator, usually a cyclotron, is used to produce a beam of protons having sufficient energy to surmount the Coulomb barrier of a particular target nucleus.

An example of this kind of reaction is $^{68}Zn\ (p, 2n)\ ^{67}Ga$. In this case the target nucleus ejects two neutrons immediately after absorbing the incident proton.

$^{203}_{81}Tl\ (p,3n)\ ^{201}_{82}Pb \xrightarrow{EC} {}^{201}_{81}Tl$

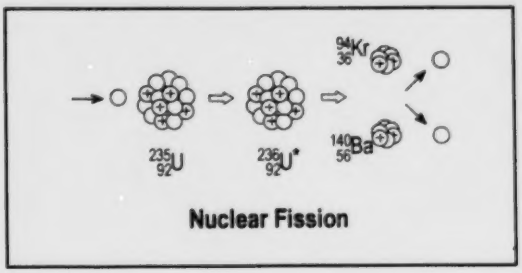
$^{130}_{52}Te\ (n,\gamma)\ ^{131}_{52}Te \xrightarrow{\beta^-} {}^{131}_{53}I$

$^{124}_{52}Te\ (p,2n)\ ^{123}_{53}I$

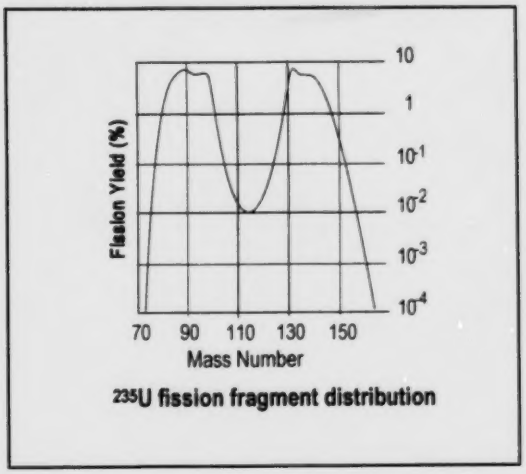
$^{112}_{48}Cd\ (p,2n)\ ^{111}_{49}In$

Radionuclide production reactions

Another source of radionuclides is **fission by-products**. A U-235 nucleus (for example) can be induced by neutron capture to split into two almost equal parts, i.e. to undergo nuclear fission. This, of course, is the basic mechanism of the atom bomb and the nuclear reactor.



U-235 nuclei do not produce exactly the same two fission fragments each time they split: a probability distribution determines what is likely to be produced and what is not. As a result, nuclear fission produces a variety of nuclides, most of which are unstable, and which decay to other nuclides. Chemical separation techniques can "harvest" this mixture for useful radionuclides. This, for example, is a way of producing Xe-133, or an alternate means for Mo-99.



Obviously, many practical details have been omitted from the above discussion; the design of nuclear reactors to produce neutrons or of particle accelerators for fast protons are subjects in

themselves. In addition, target selection and preparation, and chemical or physical separation of the desired nuclide has to be carefully considered. It is often particularly important to be able to separate the newly formed nuclide from its untransformed parent (the **carrier**) to produce **carrier-free** material. To this end, a nuclear reaction is often chosen which produces a change in proton number so that the daughter nuclide is a chemically separable isotope of the parent.

A1.9 ACTIVITY

We have seen that in unstable nuclei, nucleons undergo transformations: neutrons change into protons and vice versa. These primary transformations are still often called "disintegrations".

If we have a sample of radioactive material (a "source") in which such transformations are occurring at the rate of one per second, that source is said to have an **activity** of 1 **becquerel**. The becquerel (Bq) is the S.I. unit of activity, equal to 1 disintegration per second (1 dps).

1 Ci = 37×10^9 dps = 37 GBq

1 mCi = 37×10^6 dps = 37 MBq

1 μCi = 37×10^3 dps = 37 kBq

1 nCi = 37 dps = 37 Bq

An older unit of activity is the curie (Ci), originally defined as the activity of one gram of radium, and now defined as 37 billion disintegrations per second (3.7×10^{10} dps).

1 Bq = 1 dps

1 kBq = 10^3 dps = 27 nCi

1 MBq = 10^6 dps = 27μCi

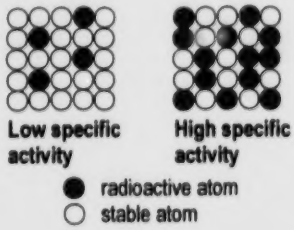
1 Gbq = 10^9 dps = 27mCi

The becquerel is a small unit, whereas the curie is a large unit, and the usual S.I. prefixes are used to create more practical sub-units.

In radiation protection, we are often concerned with the intensity of radiation emitted by a source. A high activity source will emit more radiation per second than a low activity source, but the precise relationship between activity and radiation intensity depends on the decay scheme of the radionuclide involved. A 1 Bq source may, for example, emit more than 1 gamma ray per second.

A radioactive sample may contain a relatively large percentage of unstable nuclei, or only a relative small percentage, with the remainder of the sample being made up by stable atoms. The total mass of a sample is no indication of its activity.

The **specific activity** of a sample is its activity per unit mass, e.g. the number of MBq per gram. In patient injections, for example, it is sometimes important to have a high specific activity material so that sufficient activity may be administered without also administering large amounts of non-radioactive material.



A1.10 EXPONENTIAL DECAY

Suppose we started out with 1000 unstable nuclei of a particular nuclide, and were able to watch them as they transformed ("disintegrated"). The number of nuclei transforming in any given second will be proportional to the number of unstable nuclei present.

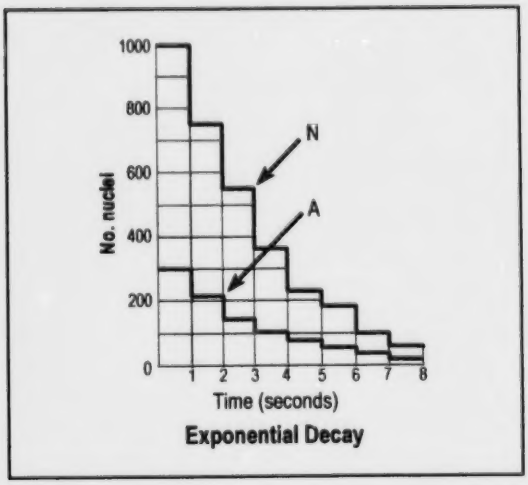
Let us suppose that in any second, 30% of the unstable nuclei present transform. So, during the first second we would see 300 nuclei transforming. During the next second, 30% of the remaining unstable nuclei would transform, i.e. 30% of 700 = 210 transformations. The table below shows what happens as we lose 30% of the remaining nuclei in each successive second.

Time (seconds)	0-1	1-2	2-3	3-4	4-5	5-6
No. nuclei (N)	1000	700	490	343	240	168
No. transforms (A)	300	210	147	103	72	50.4

Note that the final row of the table (No. of transforms) is the activity A of the source in becquerels. If we plot this activity against time, we get a characteristic curve, known as an **exponential decay curve**, because it can be mathematically expressed as $A = A_0 e^{-\lambda t}$, where A is the activity at time t, A_0 is the initial activity at time 0, λ is the "decay constant" characteristic of the radionuclide involved, and e is a number (2.7183), which is the basis of the natural logarithms (symbol ln).

The number of unstable nuclei N remaining after time t follows a similar formula:

$$N = N_0 e^{-\lambda t}$$



After a certain time, $T_{1/2}$, the number of unstable nuclei has fallen to half the original number, and at the same time the activity has fallen to half its original value. This time is called the **"half-life"** of the radionuclide. In our previous example the half life was about 2 seconds, and for the radionuclide Tc-99m it is 6 hours.

The **"mean life"** of the radionuclide is simply the mean or average lifetime of an unstable nucleus before it transforms, and it can be shown that the mean life, $T_m = 1/\lambda$, and that

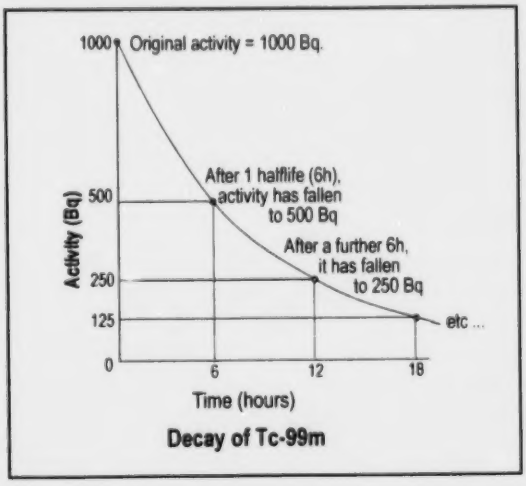
$$T_m = 1.443\ T_{1/2}.$$

In radiation protection, half life is more commonly used than mean life. Appendix XA1 contains the mathematical details of exponential decay.

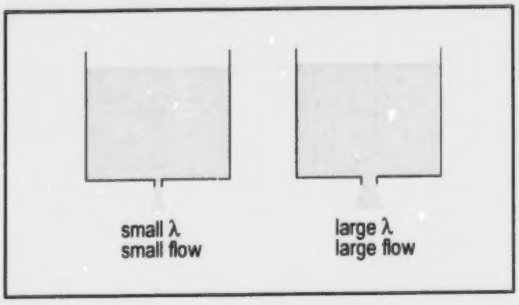
We should note that there is nothing special or unique about the exponential nature of radioactive decay: it comes quite naturally from the fact that the number of transformations per second at any given time is always proportional to the number of unstable nuclei available at that time.

A1.11 COMPLEX DECAYS

Before proceeding to more complex cases, let us describe the simple exponential decay in terms of



a popular analogy, that of water pouring from a container. The initial volume of water represents the initial number of unstable nuclei. The water pouring out of the nozzle represents the activity - the transformation and removal of radioactive nuclei. The size of the nozzle is determined by the decay constant λ ; a large value of λ means a large nozzle, a high activity and a fast rate of decay.



We also have to assume that the rate of efflux is proportional to the amount of water in the tank (activity proportional to number of nuclei). This could be achieved in a real experiment by using a specially shaped tank, but for our analogy, a rectangular tank will do.

What would happen if the unstable nuclei had not one, but two or three possible decay modes? An example is Br-80, which can decay to Kr-80 by negatron emission (39%), or to Se-80 by positron emission (19%) or electron capture (42%). The "partial" decay constants for these three branches are: $\lambda_1 = 0.021\ h^{-1}$, $\lambda_2 = 0.010\ h^{-1}$, and $\lambda_3 = 0.023\ h^{-1}$. The "total" decay constant is simply the sum of the three partial constants: $\lambda_1 + \lambda_2 + \lambda_3 = 0.054\ h^{-1}$.

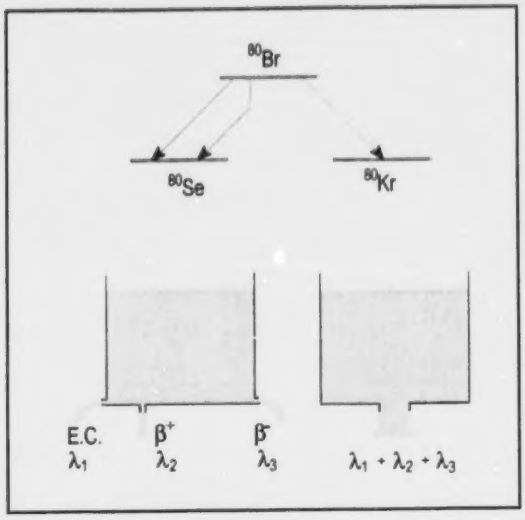
Using the container of water analogy, the container has three nozzles of different sizes, and the rate of outflow of water is as if it had one large nozzle equal in size to the sum of the three smaller ones.

In appendix XA1, it is shown that the decay constant λ and half life $T_{1/2}$ are related by $\lambda = \ln 2/T_{1/2}$. It follows that the total or "effective" half life T_E can be deduced from the total decay constant:

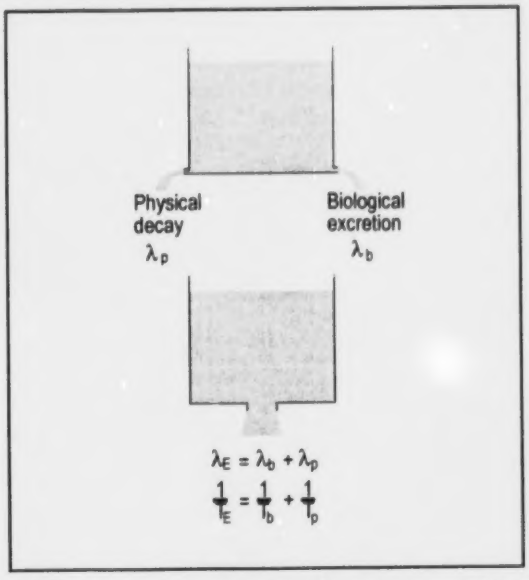
$$\lambda = \lambda_1 + \lambda_2 + \lambda_3$$

i.e. $\ln 2/T_E = \ln 2/T_1 + \ln 2/T_2 + \ln 2/T_3$

i.e. $1/T_E = 1/T_1 + 1/T_2 + 1/T_3$



When a radionuclide is injected into a patient, or accidentally ingested by a member of staff, two simultaneous processes begin to remove it: **physical decay** and **biological excretion**.



Biological excretion is a complex process which is usually analyzed by "compartmental modelling" techniques, but a rough approximation of the

removal of a radionuclide from the body can be obtained by assuming it follows a simple exponential disappearance, with a biological half life T_b and a biological decay constant λ_b. At the same time, the radionuclide is being removed by physical decay, with a half life T_p and decay constant λ_p. The combined effect of the two removal mechanisms can be described as a single exponential decay with an effective decay constant $\lambda_E = \lambda_b + \lambda_p$ and an effective half life T_E, where $1/T_E = 1/T_b + 1/T_p$.

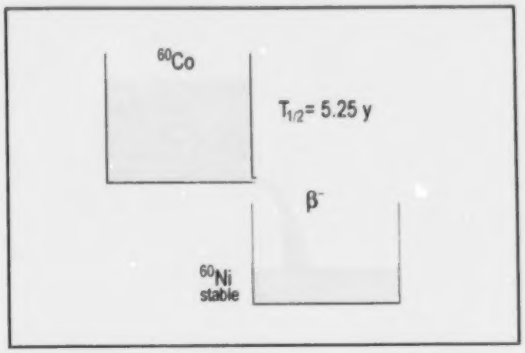
Obviously, the addition of biological excretion produces a more rapid removal of the radionuclide (a shorter effective half life) than physical decay alone. Indeed, for some long lived radionuclides, biological excretion may be the only significant removal process, in which case the effective half life equals the biological half life. On the other hand, some radionuclides are excreted very slowly or not at all. In this case, the effective half life is determined largely by physical decay.

The most hazardous radionuclides so far as internal contamination is concerned, are those having a long biological half life and a long physical half life. Examples include Sr-90 and Ra-226 which are incorporated into the bone (**bone seekers**) and have physical half lives of 28 years and 1600 years respectively.

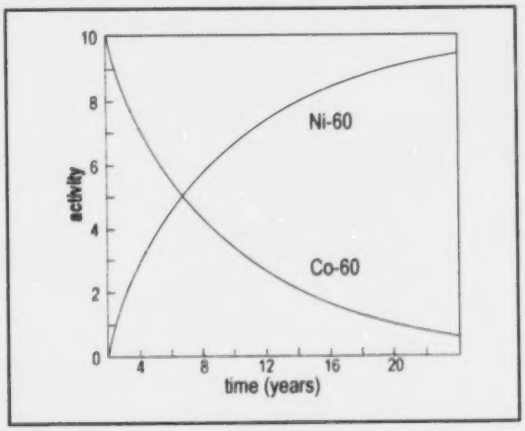
A1.12 SERIES DECAY

There are many instances where one unstable nuclide decays to a second unstable nuclide, which may in turn decay to a third, and so on. The naturally occurring radioactive series are examples of this **series decay.** Using our water container analogy, the first container drains into a second container, which begins to fill, and at the same time, begins to drain into a third container. Various situations can develop depending on the relative half lives of the parent, daughter, and granddaughter.

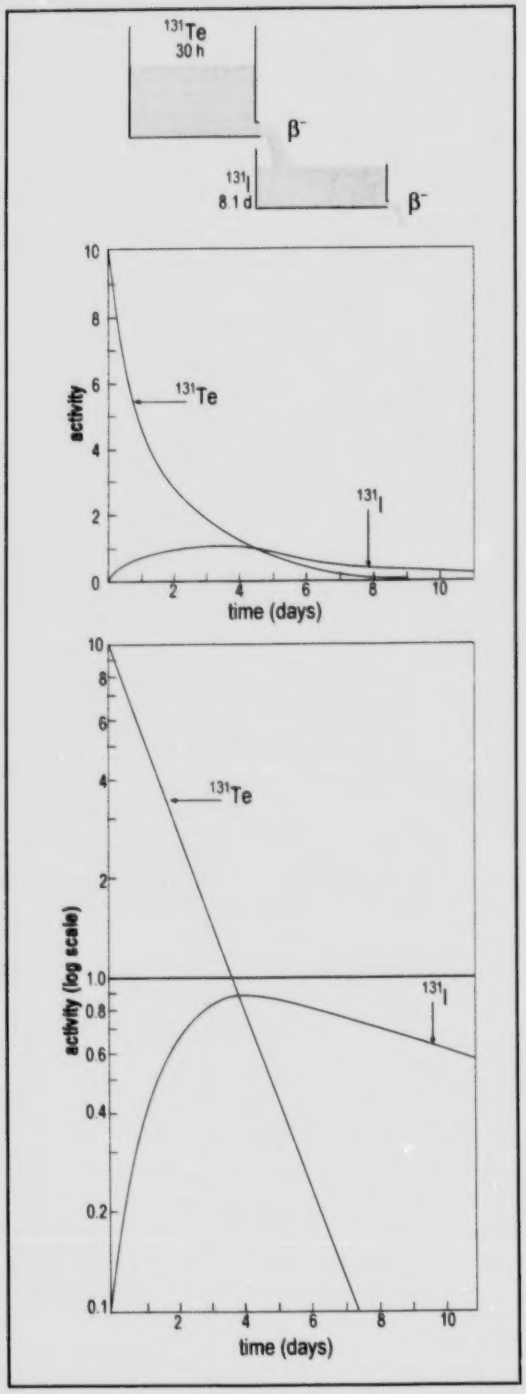
Our first example - the simplest - is where the daughter nuclide is stable; i.e. the second container has no outlet. In this case the second container fills at exactly the same rate as the first container empties; i.e. the amount of daughter nuclide increases at the same rate as the parent decreases.



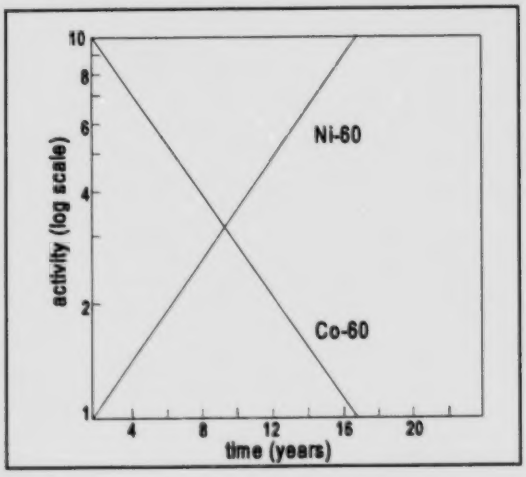
Note that if we plot the activity or number of remaining nuclei on a logarithmic scale, our exponential curve becomes a straight line.



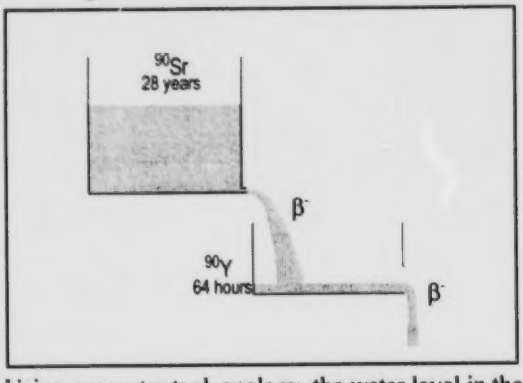
The next example is that of a short lived parent decaying to a relatively long lived daughter. In this case the parent container quickly empties into the daughter container, which fills up and then decays with its own slow half life. The first part of the activity graph shows the rapid fall of the parent and the simultaneous build up of the daughter. When the parent has all gone, the daughter decay becomes pure exponential with its own characteristic half life - a straight line on the log/linear scale.



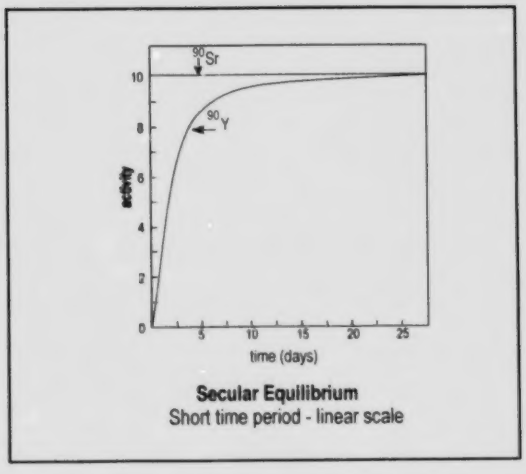
Finally we have the situation where the parent has a longer half life than the daughter. As the parent transforms, the number of daughter nuclei increases from zero. As the number of daughter nuclei increases, the rate at which they decay (their activity) also increases, until a point is reached where they are decaying at the same rate as they are being produced. Since they are being produced at



the rate the parent is decaying, they decay at the same rate as the parent decays. A kind of "equilibrium" is thus established where the parent and daughter activities are equal.



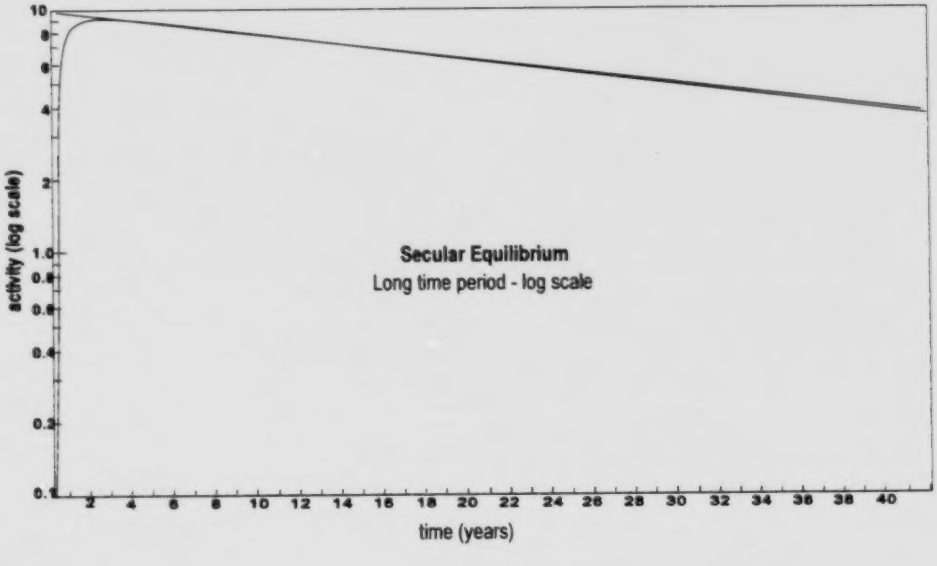
Using our water tank analogy, the water level in the daughter tank does not even cover the whole outlet, so that the outflow equals the inflow, and is not influenced by the size of the daughter outlet.



Secular Equilibrium
Short time period - linear scale

equilibrium ("secular" means "continuing over vast periods of time")

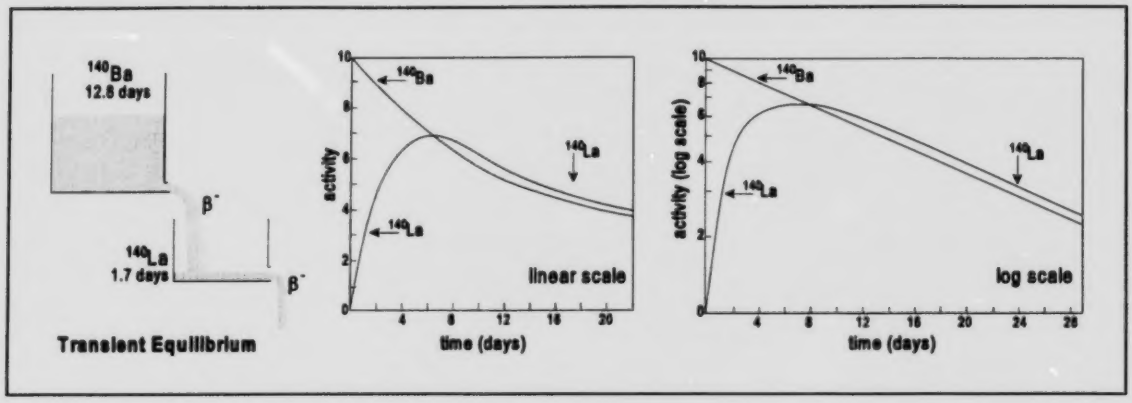
When the parent half life is only ten or a hundred times greater than the daughter's, essentially the same situation exists, but some features which appear insignificant on the secular time scale, become more apparent on the shorter time scale. This situation is known as **transient equilibrium** ("transient" means "not lasting").



Secular Equilibrium
Long time period - log scale

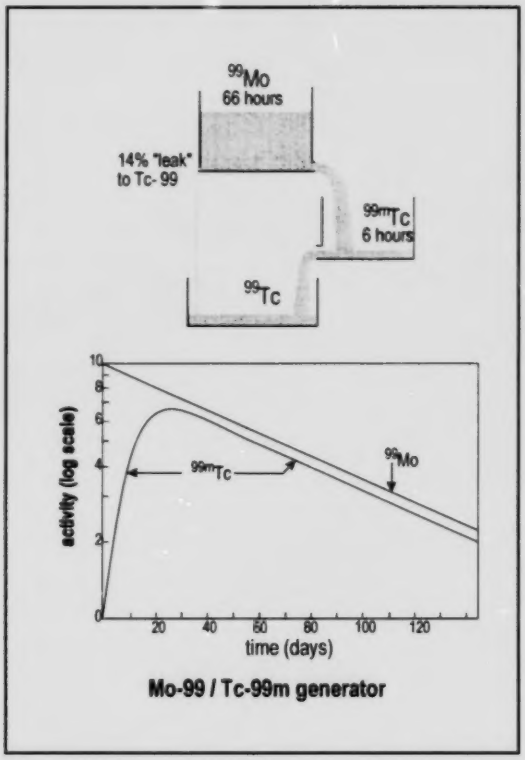
When the parent half life is millions of times greater than the daughter's, the parent activity appears to be constant on a time scale chosen to show the rise of the daughter. The daughter activity initially climbs from zero until it equals that of the parent, and then follows the parent. This situation has been given the title of **secular**

The rise of the daughter occupies a greater portion of the observation time, and the period during which the parent and daughter activities run parallel occupies less time. It also becomes apparent that the parent and daughter activities are not equal, but the daughter activity "overshoots" the parent activity. The quickest way to understand



this without complex mathematics is to realize that the area under the parent decay curve, integrated to infinite time, is the total number of parent nuclei. If every parent nucleus produces a daughter nucleus, we must eventually end up with the same number of daughter nuclei, which is the area under the daughter decay curve. Since the parent curve starts off higher than the daughter curve, the daughter curve must become higher than the parent curve if the areas under the two curves are to end up equal.

The fact that the short-lived daughter is available in an amount which diminishes with the longer half life of the parent is the basis of the **radionuclide generator**, the most common example of which is the Mo-99/Tc-99m generator. In many cases, not all parent nuclei produce the desired daughter. For example, 14% of Mo-99 nuclei decay immediately to the ground state of Tc-99, bypassing the metastable Tc-99m state- a "leak" in the parent container. Thus, the activity of Tc-99m in a generator does not overshoot the parent Mo-99, but rises to only about 95% of it.



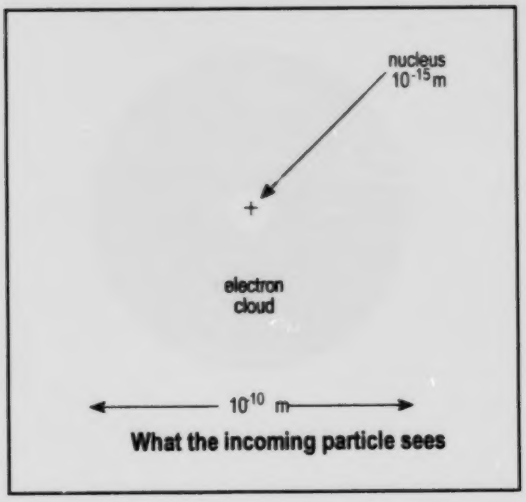
CHAPTER A2

RADIATION INTERACTIONS WITH MATTER

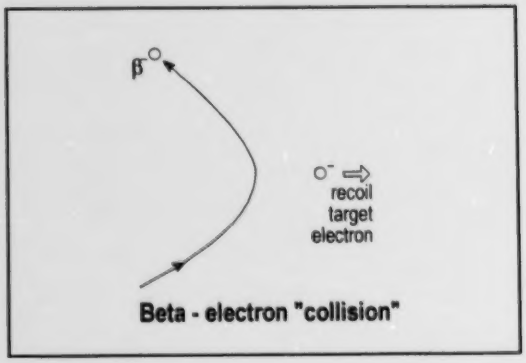
A2.1 INTRODUCTION

Before discussing what happens when a moving "sub-atomic" particle, such as an alpha or beta particle, strikes a material object (the target), let us recall a few features of the atoms of which the target is composed.

The electron orbits of an atom account for most of its size. The diameter of the "electron cloud" is about 10^{-10}m, while the nucleus is about 10^{-15} m. If the nucleus was the size of a garbage can (say 1m), the nearest electron would be a fly orbiting about 100 kilometers away. Incoming particles therefore, interact mainly with the electrons surrounding the nucleus. We will mention some exceptions where the incoming particle may by-pass the electrons and interact with the nucleus.



Although we often talk of particles "hitting" electrons, it is the extensive electric field generated by an incoming charged particle that collides with the extensive electric field generated by the electron. Some high-school physics labs have repulsive magnetic disks riding on air cushions which stimulate such interactions by "colliding" without touching.



Bearing this in mind, we shall continue to treat particle/electron interactions like balls on a pool table, often referred to as "billiard ball collisions" from an English version of the game.

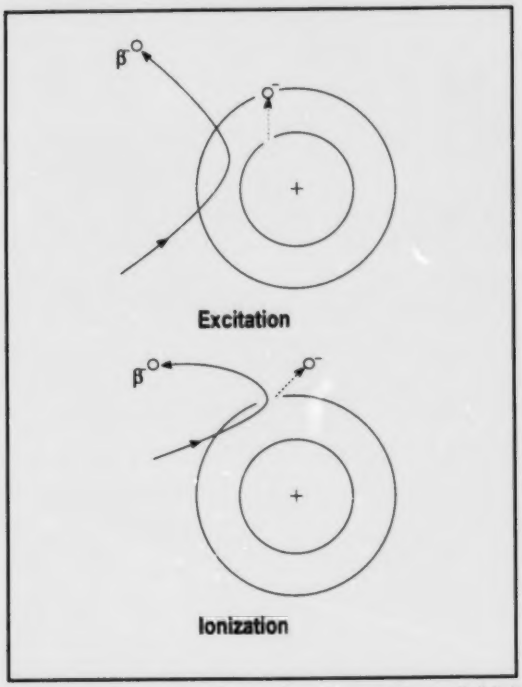
However we visualize the interaction, the outcome is that the incoming particle transfers some of its energy to a target electron. The amount of energy transferred depends on whether the incoming particle happens to hit the electron "head on" (large energy transfer) or off to one side.

Basically, one of three things can happen, depending on how much energy is transferred:

1. nothing happens - if the amount of energy available for transfer is less than that needed to raise the target electron to the next available energy level, no energy is transferred;

2. **excitation** - the target electron receives enough energy to raise it to a higher "excited" level;

3. **ionization** - the target electron receives so much energy that it is excited beyond the highest level, i.e. it pops out of the "hole" and leaves the atom altogether. The detached electron is a "negative ion", and the remaining atom is a "positive ion".



Some electrons may leave the atom with such high velocities that they can produce excitations or ionizations in neighbouring atoms. These "new beta particles" are called "delta rays".

Let us pause to note that there are many kinds of "radiation" in our world: we talk of sound radiation, of heat radiation (electromagnetic radiation in the infrared part of the spectrum), ultra-violet radiation, micro waves and radio and so on. Some of these radiations carry enough energy to produce excitations in target atoms, but only x-rays, gamma rays and the various sub-atomic particles mentioned, carry enough energy to produce ionizations in target atoms. The popular press sometimes calls this "atomic" or "nuclear" radiation, but the proper term for radiation which can produce ionization is quite simply "**ionizing radiation**".

We can divide the various kinds of ionizing radiation into two main categories:

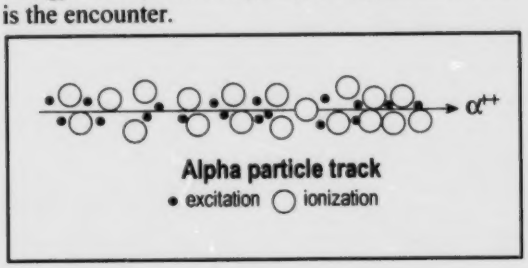
1. directly ionizing radiation - this consists of charged particles such as alpha or beta particles, which interact with the target electrons via the Coulomb electric force, and

2. indirectly ionizing radiation - this includes neutral particles such as neutrons, and high energy photons such as X and gamma rays.

A2.2 CHARGED PARTICLES

An **alpha particle** is about 7000 times more massive than an electron, and carries two units of positive charge to the electron's single negative one. If we imagine the target electrons to be ping-pong balls, then the incoming alpha particle is a ten-pin bowling ball.

As the alpha particle passes atomic electrons, its attractive Coulomb field "sucks" the electrons from their atoms (ionization), or merely into a higher energy level (excitation), depending on how close is the encounter.



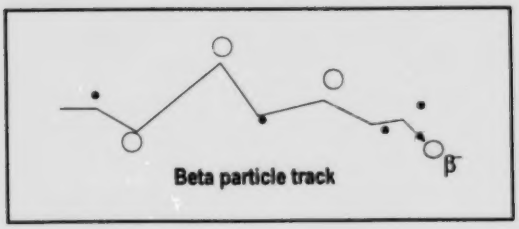
Because the alpha particle is so massive, a single "collision" with an electron has little effect on its course, and it ploughs a straight path through the target electrons. (On rare occasions, a target nucleus may happen to lie in its path, and a more violent collision will occur, with a drastic change in trajectory resulting. It was such rare collisions which lead Rutherford to "discover" the atomic nucleus.)

The momentum transferred from the alpha particle to an electron is given by the "impulse", which is the force between them multiplied by the time during which the force is exerted.

So the closer the alpha approaches the electron, the greater the force, and the greater the momentum (and energy) transferred. Also, the greater the time the alpha spends close to the electron, the longer the force lasts, and the greater the impulse and energy transferred. Thus, a slow alpha, which takes longer to pass the electron, can transfer more energy than a fast one.

As the alpha particle loses kinetic energy a little at a time to the target electrons, it slows, and therefore begins to lose increasing amounts of energy with each collision. The increased energy transfer produces increased ionization. Detectors, such as cloud chambers and photographic plates, which reveal the ionization tracks of charged particles, show this effect as an increased "ionization density" towards the end of an alpha particle track.

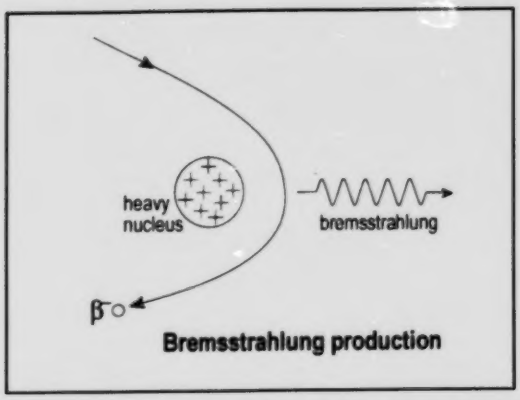
A **beta particle** has, of course, the same mass as the target electrons and a single collision with an electron can produce a drastic change in the beta's course. Rather than plough a straight path like the massive alpha particle, a beta will be bounced from electron to electron in a random zig-zag path. Like the alpha particle, it produces excitations and ionizations along its path.



Since an alpha particle has about 7000 times the mass of a beta particle, and since kinetic energy is given by $E = {}^1/_2\, mv^2$, an alpha particle moves more slowly than a beta particle of the same energy. This, coupled with the alpha particle's double charge, means that an alpha particle produces a heavier, denser ionization track than a beta particle. From a radiation safety point of view, the important differences between alpha and beta particles are : alphas are more heavily ionizing than betas, and are less penetrating.

In addition to the Coulomb interaction with electrons, there is another mechanism by which a beta particle can lose energy. As previously mentioned, when a moving charged particle is suddenly slowed or has its direction changed (i.e. is accelerated), it emits electromagnetic radiation at a rate proportional to the square of the acceleration. This is true for all charged particles, but "heavy" particles like alphas are too massive to undergo high accelerations, and under normal circumstances, only beta particles are light enough to show this effect.

If a beta particle comes close enough to a (positive) nucleus, the attractive Coulomb field will swing it around onto a new course. In the process, the electron emits electromagnetic radiation in the form of x-rays. This x-radiation is known as **bremsstrahlung**, which is German for "braking radiation", and has been likened to the heat and noise radiated by a fast-cornering car.



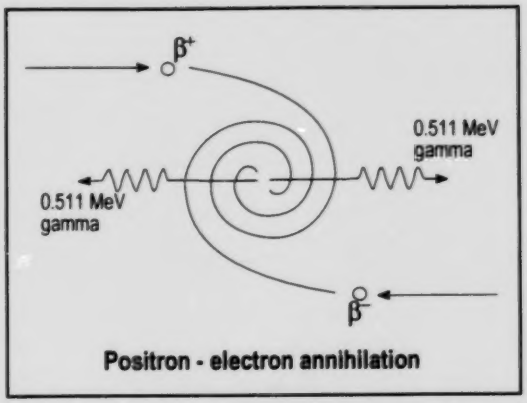
Bremsstrahlung production

Bremsstrahlung are (mainly) what comes out of x-ray machines, and are generated by boiling off electrons from a filament, accelerating them through a high voltage, and then decelerating them quickly in a heavy target.

Bremsstrahlung emission is more pronounced if the target nuclei have a high atomic number, since the greater nuclear charge produces a stronger Coulomb force and a greater deceleration in the beta particle. Bremsstrahlung emission also increases with increasing beta energy and only becomes significant at beta energies above 1 MeV.

Thus, to avoid bremsstrahlung production, radionuclides which emit only high energy betas (pure beta emitters, such a P-32), are better stored in a low-Z plastic container, such as Lucite, rather than the lead pot normally used for storing gamma-emitting nuclides.

Positrons interact with target electrons in much the same way as beta particles. The Coulomb force is attractive, rather than repulsive, but the strength of the interaction and the masses of the two particles are the same. However, when the positron has been slowed enough, it will be "captured" by an electron. The electron and positron will orbit around a common centre of gravity, spiralling inward until they meet, when they annihilate each other.



Positron - electron annihilation

This is a classic example of matter meeting anti-matter, and results in the total conversion of the mass of the two particles into energy in accordance with the equation $E = mc^2$. For an electron/positron pair, the annihilation radiation produced consists of two gamma rays, each of 0.511 MeV of energy. The detection of these gamma rays forms the basis for positron emission tomography (PET).

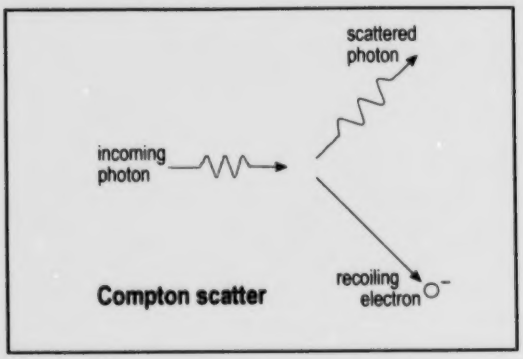
It should be noted that adequate lead shielding is required to shield positron emitting radionuclides.

A2.3 PHOTONS

It is usually convenient to treat low energy electromagnetic radiations such as radio and visible light, as waves. However, to explain the interaction of the higher energy x and gamma radiations, their quantum or particle nature must be emphasized, i.e. they are treated as particles called photons moving at the velocity of light (3×10^8 m/s).

There are three different mechanisms by which such high energy photons interact with target atoms: Compton scattering, the photoelectric effect, and pair production.

In **Compton scattering**, the incoming photon "bounces off" an electron and continues on in a new direction with reduced energy. This "scattered" photon may subsequently interact with another target electron.



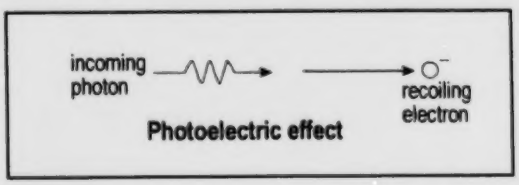
The energy lost by the original photon is transferred to the target electron, which leaves its parent atom and itself becomes an ionizing particle.

As before, the amount of energy transferred depends on whether the photon collides almost "head on" with the electron, or merely strikes a glancing blow, causing a small change in the photon's direction.

Compton scattering is most likely with electrons that are loosely bound to their parent atom, (sometimes called "free" electrons) i.e. outer shell electrons of low-Z targets. It is also more likely at lower photon energies.

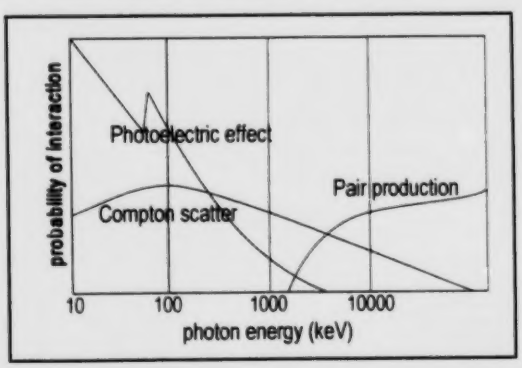
Tightly bound electrons (inner shell, high-Z targets), are more likely to participate in the second process, the **photoelectric effect**. This is like a "head on" snooker shot, where the cue ball comes to rest, and the target ball shoots off at high speed. Since photons cannot come to rest (they always travel at the speed of light), the photon disappears, and the target electron shoots off with all the energy the photon had, minus the comparatively small amount used up in detaching the electron from its parent atom (the "ionization energy").

The third process, **pair production**, is the reverse of the electron/positron annihilation described before: the photon energy is used to create an electron and a positron. Using $E = mc^2$, it is easy to calculate that 0.511 MeV of energy is needed to produce the mass of an electron (9.11×10^{-31} kg), and of course, the same amount for a positron. Pair production, therefore, cannot occur unless the original photon carries at least 1.022 MeV of energy. Any photon energy left over after the mass creation, is divided randomly between the electron and positron.



The process also requires the presence of a strong electric field, such as is found close to a high-Z nucleus, and is thus more common in heavy targets. Above the 1.022 MeV threshold, the probability of pair production increases rapidly with increasing photon energy.

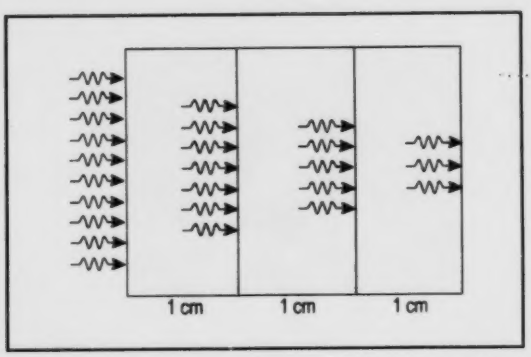
The relative importance of each of the three photon interaction processes depends on the target material and the energy of the photons. This relationship between energy and probability of interaction is illustrated in the accompanying graph.



A2.4 ATTENUATION OF PHOTONS

The interaction of an x or gamma ray photon with target electrons is less likely, and more of a "hit or miss" process than the interaction of a charged particle. A charged particle, such as a beta particle, is surrounded by an electric field whose strength decreases with the square of the distance from the particle. A beta particle may "miss" a target electron by a good margin and still lose a little energy to it because of the interaction of the electric fields.

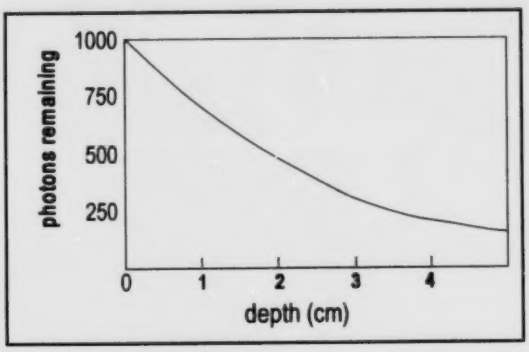
A photon, on the other hand, either "hits" the target electron, and loses much or all of its energy, or it "misses" and carries straight on with no energy loss. Because of a photon's lower probability of interacting, sometimes referred to as a lower "interaction cross-section", x and gamma radiation is more penetrating than charged particle radiation. Because of the "all or nothing" nature of the photon interaction, the way it penetrates a material is also different.



Suppose a beam of gamma radiation initially contained 1000 photons, and suppose that for every centimetre the beam travelled in a material, 30% of them were lost from the beam due to interactions with target electrons. After the first cm into the target, there would be 700 photons left in the beam. During penetration of the next cm of target material, 30% of these would be lost, i.e. 210, leaving 490. And so on, as shown in the following table.

Depth into material	Photons lost	Photons remaining
0 cm	0	1000
1 cm	300	700
2 cm	210	490
3 cm	147	343
etc		

A plot of beam intensity (i.e. the number of photons remaining) as a function of the depth into the target material, produces the curve shown.



The mathematics of this process is identical to that used to describe the exponential decay of a radionuclide, except that it is thickness of target material instead of time, that reduces the radiation intensity. The intensity I at any depth x is given by $I = I_0 e^{-\mu x}$, where I_0 is the initial intensity.

The quantity μ (which corresponds to the decay constant λ) is known as the **"linear attenuation coefficient"** for the material.

In radionuclide decay, a certain time, called the half life, reduces the activity to half its original value. In photon attenuation, a certain thickness of material, called the **"half value layer"** or HVL, reduces the radiation intensity to half its original value.

Higher energy photons tend to be more penetrating than lower energy photons, and heavy elements, such as lead, with numerous electrons per atom, are more effective at stopping photons than light elements. Both these facts are reflected in the HVL value which is quoted for a particular material at a particular photon energy. Some common examples are given below.

Half Value Layers (HVL) of some common materials (in cm)			
Photon energy	Lead	Aluminum	Water
20 keV	0.0007	0.0747	0.975
140 keV	0.0256	1.796	4.530
364 keV	0.2190	2.646	6.245

Thus, knowing the HVL value for a shielding material, it is possible to estimate the thickness required to reduce photon radiation to a safe level.

For radiation safety purposes, the following consequences of neutron interactions are important:

1. neutrons can make a target material radioactive.
2. neutrons are best stopped by materials containing hydrogen nuclei, e.g. water, paraffin wax.
3. neutrons can produce reaction products which cause high density ionizations. The biological effects of neutron irradiation are complex, and less well understood than those of beta or gamma rays.

Many other kinds of ionizing radiation exist, mostly due to human intervention. These include beams of protons, deuterons, mesons, etc., none of which is relevant to hospital radiation protection.

A2.5 ENERGY TRANSFER

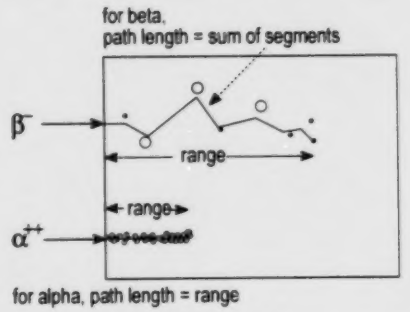
At this point, we need to define some concepts and terms which will be used in chapters A4 and A6.

Be aware that these are approximations to the more precise definitions used in the detailed, mathematical analysis of radiation interactions. Such precise definitions are regularly revised and published by an organization called the International Commission on Radiological Units and Measurements or ICRU (see chapter A7).

Path Length and Range:

The **path length** of an individual charged particle is the total distance it travels, irrespective of direction.

The **range** for a charged particle is the average path length for many identical particles, all having the same initial energy, projected in the original direction of incidence of the particle. For our purposes, range is the same as depth of penetration.



Path length and range

One might deduce from the above definition of range that identical particles of identical energy do not necessarily travel the same distance. This variation in path length is called **path length straggling**, and is due to the random or statistical nature of collisions with target atoms.

Stopping Power:

A charged particle comes to a stop when it has lost its kinetic energy. The **stopping power** of a target material is the energy loss per unit path length in the material, often expressed in MeV per cm.

We have seen that charged particles lose energy to target electrons through the "collision" of their electric fields, and indeed, energy lost by this mechanism is called "collision loss".

We have also seen that charged particles, most especially electrons, can also lose energy by the bremsstrahlung process. This type of energy loss is referred to as "radiative loss".

Total stopping power is the sum of "collision stopping power" and "radiative stopping power". Energy lost by the radiative process may be deposited in the target some distance from the main particle track, so that when we are concerned with the local deposition of energy, we often use only the collision component.

Linear Energy Transfer (LET):

Linear Energy Transfer, L_{Δ}, is a restricted collisional stopping power. By restricted we mean that we exclude energy transfers above a given energy Δ. For example, $L_{.01}$ for 10 keV electrons in water is 1.426 MeV/cm, which means that the electrons lose an average of 1.426 MeV per cm of water traversed, if we ignore any interaction which results in an energy loss greater than 0.01 MeV.

The reason for this restriction is that high energy transfers produce "delta rays", fast moving target electrons which carry off energy beyond the local area of interest. Thus, Linear Energy Transfer is an indication of how much energy the particle deposits locally, an indication of the "ionization density". High LET particles, such as alpha particles, produce a higher ionization density than low LET particles, such as beta particles.

This is important because the amount of irreparable biological damage produced increases with ionization density.

Strictly speaking, the concept of LET is meaningful only for charged particles, but it is often applied also to photons, since they interact to produce ionizing charged particles (i.e. electrons).

Kerma:

For indirectly ionizing particles, such as photons or neutrons, energy transfer to the target material takes place in two stages:

1. the uncharged particle produces a charged particle or particles and transfers energy to it or them (e.g. a photon produces a photoelectron);

2. the charged particle(s) transfer energy to the target material (e.g. the photoelectron produces excitations or ionizations in the target atoms).

Kerma, an acronym for Kinetic Energy Released per unit Mass, is the sum of the kinetic energies of all the charged particles produced in the first stage mentioned above, per unit mass of target material.

A2.6 Sources Of Radiation Exposure

Radiation is a natural part of life. It has existed since the beginning of time and is an integral part of the universe in which we live. Life as we know it on earth has evolved in the presence of radiation. Radiation comes to us from many sources both natural and man-made, and in Canada we are all exposed to about 2 mSv per year.

The following table gives a summary of the contribution from these various sources of radiation.

Radiation Source	Annual Effective Dose (mSv)
Natural	
Cosmic radiation	0.30
Gamma rays from the earth	0.35
Internal sources	0.35
Radon	1.00
Man-made	
Medical	1.50
Fallout from weapon tests	< 0.01
Nuclear power	< 0.001
TOTAL	3.5

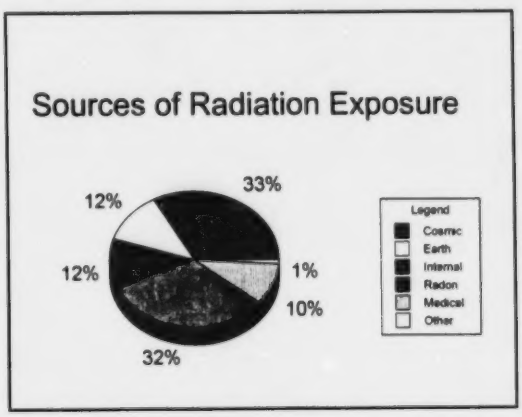
Most of the radiation exposure of the general public comes from natural sources. Roughly a third of a millisievert per year comes from each of the three sources - cosmic radiation from outer space, radiation from the soil and buildings and natural isotopes in our own bodies.

Cosmic radiation and terrestrial radiation vary with location. For example, the radiation dose from cosmic radiation increases with altitude. At an altitude of 3000 metres the annual rate would be increased by 0.5 mSv. Travelling by air increases radiation dose because of altitude also. On a one-way trip from Halifax to Vancouver an extra 0.02 mSv would be received.

Terrestrial radiation varies with the type of soil and building materials from which buildings, especially houses, are constructed. A house made of brick increases the dose by 0.4 mSv and for a house made of granite the dose would be increased by 1.2 mSv.

The largest source of radiation exposure from the environment is through the inhalation of radon. Radon-222 is a radioactive inert gas which is formed as a decay product from naturally occurring uranium and Radon-220 is a decay product of thorium, both common elements in the soil. It diffuses out of the soil and into our homes and other buildings. This source of radiation exposure varies widely across Canada, depending mainly on the concentration of radioactive elements in the soil, from a low of 0.2 mSv in Vancouver, 0.4 mSv in Montreal, 0.55 mSv in Toronto, to 2.2 mSv in Winnipeg.

In hospitals, radiation is an important tool in the diagnosis and treatment of disease, and modern medicine would virtually come to a stop without its use. It is not surprising therefore that the largest man-made source of radiation exposure is from medical x-rays. A reliable estimate for the average annual exposure of the Canadian population is not available although it is probably similar to other



developed countries and of the order of 1.0 mSv.

(UNSCEAR 1993)

CHAPTER A3

RADIATION DETECTORS

A3.1 INTRODUCTION

As discussed in Chapter A2, ionizing radiation is so named because it interacts with the atomic electrons of a target material to produce negative and positive ions. The operation of almost all radiation detectors depends, in one way or another, on the collection and sometimes the quantitation of these negative and positive ions.

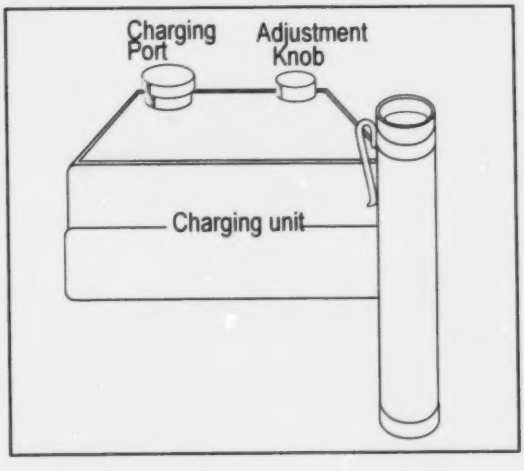
Numerous gases, liquids and solids have been used as the target material for detectors, and a variety of methods developed for transforming the ionization into a readable measure of the incident radiation, so that a wide range of different instruments is available for different circumstances and applications.

Some of the more common of these instruments with applications in radiation safety are described in this chapter.

A3.2 POCKET DOSIMETER

This usually resembles a fat fountain pen, and may also be called a "Personnel Dosimeter", a "Quartz-Fiber Dosimeter (QFD)", or a "Direct Reading Dosimeter" (DRD).

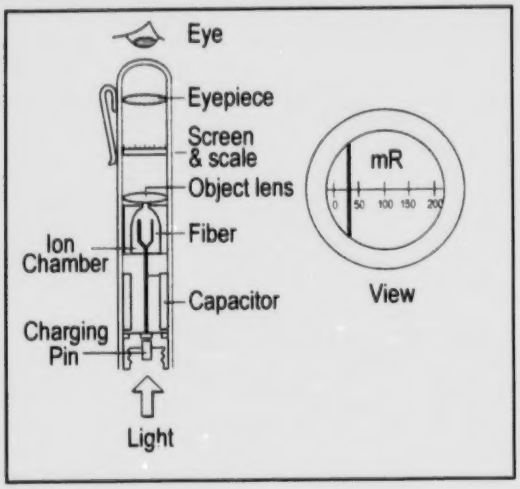
These instruments are clipped to a person's lapel or pocket to monitor their radiation exposure, and have the advantage over TLD badges (section A3.10) in yielding immediate results.



The dosimeter is charged by inserting one end into the socket of a battery-powered charging unit, causing the fiber to move because of electrostatic repulsion. During this process, a lamp in the charging unit shines light up the barrel of the dosimeter and casts a shadow of the fiber onto a screen which has a scale calibrated in exposure units. A magnified image of this screen can be seen through the eye-piece at the top end of the dosimeter. A control knob on the charging unit regulates the amount of charge deposited, so that the fiber can be set to the zero position on the scale, before removing the dosimeter for use.

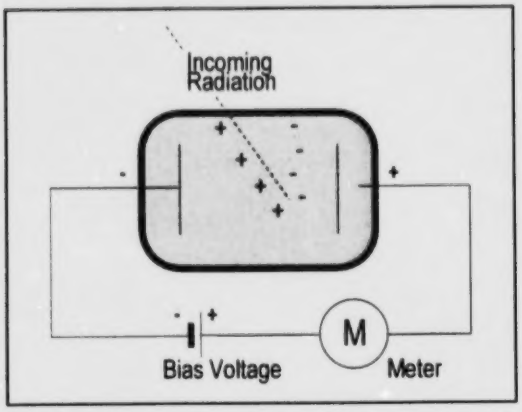
Radiation-induced ionization causes the fiber to collapse towards its original position, and the shadow registers the movement as an increase in accumulated exposure. This "self-reading" design

can be read at any time by removing a protective plastic cap, pointing the charging end to a light source, and looking through the eyepiece. Some practice may be required before the scale can be properly seen, and the reading correctly interpreted.



A3.3 Gaseous Ion Chambers

Three instruments in this general class are: the Ionization Chamber, the Proportional Counter, and the Geiger-Müller Counter. The basic principle of this class of detector is shown below.

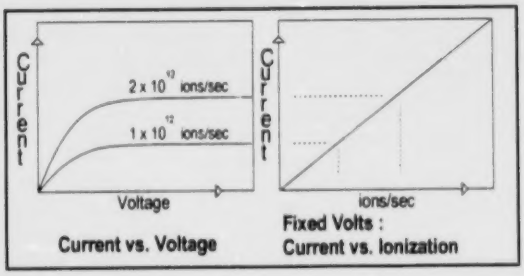


The detector consists of a gas-filled chamber with two electrodes, a positive anode and a negative cathode. Ionizing radiation entering the chamber produces negative ions (electrons) and positive ions, by interaction with the gas molecules. The negative electrons are drawn to the positive anode and the positive ions to the negative cathode. This movement of charge (an electric current) produces a response in the meter.

The detailed behaviour of the detector depends on a variety of factors, including the composition and pressure of the gas, the strength of the electric field, and the method of collection and measurement of the charge.

A3.4 Current Ionization Chamber

If the field strength (volts/cm. of electrode separation) is increased by increasing the voltage, a plot of the resulting ionization current will have the following general shape.



The first figure shows two graphs for two different intensities of radiation. For example, a radioactive source of 100 MBq (2.7 mCi) might produce 1×10^{12} ion pairs per second, and another source of 200 MBq produce ion pairs at twice that rate.

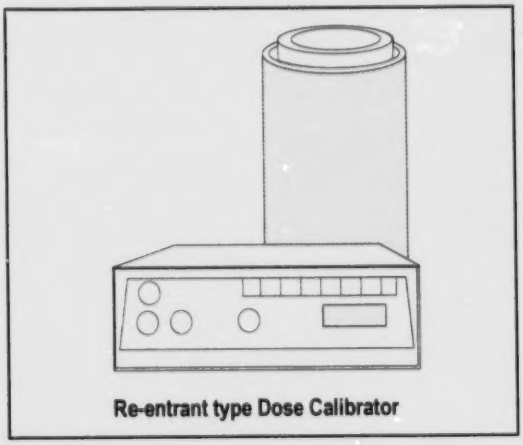
At a low applied voltage, the anode and cathode exert a weak force on the negative and positive ions, and they are only slowly separated and collected. This increases the probability that negative and positive ions will collide and recombine with each other, thus reducing the measured current. As the voltage is increased, ion separation and collection becomes more efficient, until a voltage is reached, (the saturation voltage),

at which the ions are collected as quickly as they are generated, with little recombination.

If the voltage is set at the saturation level for efficient ion collection, (several hundred volts for typical designs), then the output current (charge per second) will depend on the rate at which ions are produced, as shown in the second graph above.

The ion production rate will depend on many factors, including the type and amount (pressure) of gas in the chamber, and the type and energy of the radiation. However, the type and pressure of the gas is fixed by the design and manufacture of the instrument, and a particular radionuclide always emits the same types and energies of radiations, so that for a given instrument and radionuclide, and fixed source-detector geometry, the output current will be depend only on the source activity.

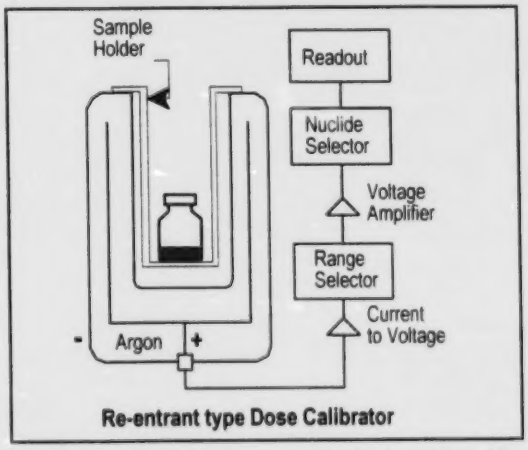
This is the basis of the radionuclide calibrator (U.K.) or activity calibrator (U.S.), which is unfortunately most frequently known as a dose calibrator. (Dose as in "dose of medicine", not as in "radiation absorbed dose").



Re-entrant type Dose Calibrator

The chamber is of the so-called "re-entrant" design (the walls turn back or 're-enter' the chamber), so that the gas (typically at a pressure of 20 atmospheres of argon) nearly surrounds the radioactive sample. This increases interception of radiation emitted from the sample in all directions (isotropically), and also makes the measurement less sensitive to changes in sample position.

Details of the electronic design vary with manufacturer, but all have some means whereby the amplified output of the detector is scaled by a factor unique to each radionuclide, before being displayed as an activity value. This is necessary because the radiation, and hence the ionization current, produced by 1MBq of one radionuclide differs from that produced by 1MBq of another, according to their different decay schemes.

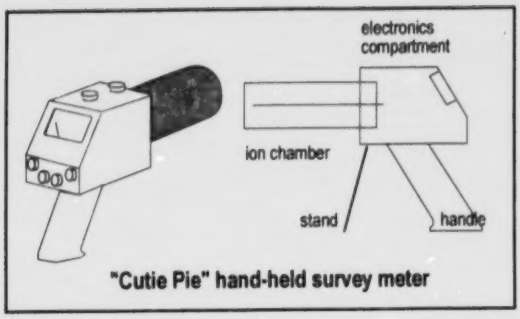


Re-entrant type Dose Calibrator

Scaling factors may, for example, be introduced by a set of resistors, each resistor value having been determined by counting known activities of the different radionuclides. The appropriate resistor is then switched into the circuit when the radionuclide selection is made by push-button, a keyed plastic insert, or similar means. Alternatively, a variable potentiometer may be used, and the radionuclide selected by setting it to the appropriate value, read from a table supplied by the manufacturer.

Note that an incorrect nuclide selection, or impurities in the sample, will defeat this system of converting output current to activity, and give inaccurate readings.

Another application of the ion chamber is in the hand-held survey meter popularly known as a "Cutie Pie".



"Cutie Pie" hand-held survey meter

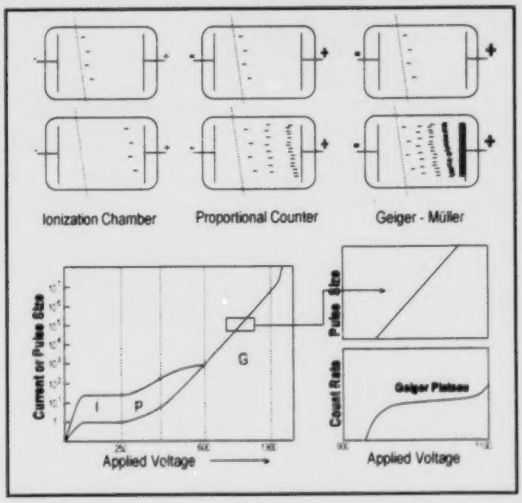
A3.5 THE GEIGER-MÜLLER COUNTER

In the radionuclide calibrator described previously, bursts of ionization due to the arrival of individual particles of radiation are deliberately smoothed out into a steady current, to provide a constant reading of source activity. For many applications, it is preferable to count the individual particles, and in the Pulse Ionization Chamber the electronics is designed to generate a sharp voltage pulse for each ionizing burst. The pulses may be counted over a set time using a "scaler/timer" to give a measure of radiation intensity (e.g. counts per minute), or converted to a steady voltage by a resistor/capacitor (RC) circuit to drive a count rate meter.

If we continue to increase the voltage applied to the electrodes of the ion chamber, beyond the saturation voltage, the ions produced by the radiation (primary ions) are accelerated to such a speed that they ionize other gas atoms with which they collide in their journey to the electrodes. The secondary ions are accelerated in turn to generate yet more ions, and so on, to produce a multiplying or amplifying effect. Up to a certain voltage, the effect remains linear, so that the total ionization produced is proportional to the original or primary ionization, and a chamber operating in this mode is a Proportional Counter.

As the voltage is increased further, the multiplying effect becomes an avalanche, assisted by the emission of photons which produce photoelectrons elsewhere in the gas, so that even the smallest amount of initial ionization results in a massive discharge. This is the operating mode of the Geiger-Müller (G.M.) tube, with an "internal amplification" factor of 10^6 or more producing an output pulse of several volts from little more than a single primary ion pair.

The following figure shows in general how the output current or pulse size from a gas ion chamber increases as the electric field is increased.

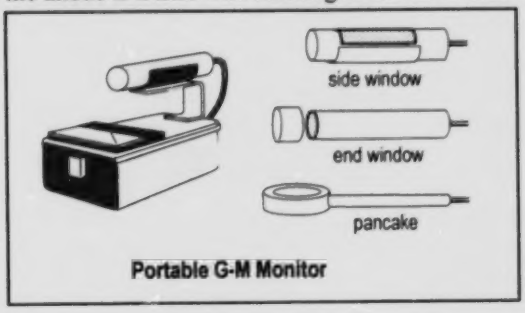


The beginning of the first graph shows two different primary ionizations. For example, the top curve might represent the heavily ionizing alpha particle, and the bottom curve, the lightly ionizing beta particle. The region labeled I is the operating range of the ionization chamber discussed in section A3.4. P is the proportional counter range, followed by a range sometimes called "limited proportionality", where the dependence on primary ionization gradually disappears. G is the Geiger range where the output current (pulse size) is independent of the initial amount of ionization, but still increases as the applied field is increased; since the Geiger Counter merely counts the pulses, regardless of their size, this does not affect the operation of the instrument. More important is the variation in the **number** of pulses, or **count rate**, which becomes relatively constant between certain voltages, as shown in the bottom right-hand graph in the figure above.

It is the strength of the electric field which determines the region of operation. Field strength depends not only on the applied voltage, but also on the separation and geometrical arrangement of the electrodes, so that the voltage values shown above are only representative, and will vary according to the detailed design of the instrument. Once a Geiger discharge is initiated by an ionizing particle it tends to be self-propagating. To improve the instrument's ability to count at high rates, the discharge is deliberately extinguished, or "quenched", either by external, electronic means, or by adding a quenching agent (e.g. chlorine) to the gas. This agent absorbs electron energies without producing secondary ions or photo-electrons and thus aborts the multiplication process.

At applied voltages above the Geiger-Müller region, the electric field itself is strong enough to cause ionization by pulling electrons from their parent atoms, and a continuous discharge results, like that in a fluorescent light.

The figure below shows a typical Geiger-Müller counter for general monitoring. The actual G.M. tube is often in the form of a cylindrical probe connected to the main instrument case by an electrical cord. The outer tube is the cathode, and the anode is a thin wire running down the centre.



Portable G-M Monitor

Batteries provide power to generate tube voltage and operate the electronics. Counts per minute can be read off a meter in several switch-selectable ranges, and a slow or fast needle response is usually available by switching in more or less electronic smoothing of the tube output.

A speaker or ear-plug announces the arrival of each ionizing particle with the clicking sound popularly associated with Geiger Counters and radioactivity in general. This audible signal is useful for contamination monitoring, when the user's eyes are fixed on the whereabouts of the probe, rather than on the countrate meter.

Three basic probe types are shown in the figure: side-window, end-window and pancake. The first two have a moveable "beta shield", usually made of aluminum. When the shield is closed, it stops charged particles from entering the window, so that the count rate represents only the gamma component of a mixed radiation field. The smaller end-window can be thinner than the side-window, and when unshielded, may permit the entry of alpha and beta particles with energies exceeding about 50 keV, whereas the thicker side window will only pass hard betas and x and gamma radiation. The pancake probe usually has a thin (Mylar) window to permit passage of charged particles, and can be used to detect surface contamination over a wider area than the other probes. G.M. probes in general have a lower sensitivity to gamma rays but a relatively high sensitivity to beta particles.

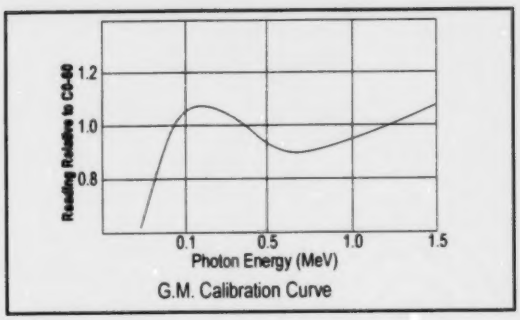
Because of its simplicity and cheapness, the G.M. counter is a popular, and sometimes the only, radiation survey instrument in radionuclide labs, and it has been adapted to read exposure as well as counts per minute.

A3.6 THE G.M. COUNTER AS AN EXPOSURE METER

Exposure (see Chapter A4.3) is the amount of ionization produced in air by photons. The G.M. counter only measures the **number** of photons, (e.g. counts per minute), regardless of the ionization produced by each one. How then can a G.M. counter be used to measure exposure in coulombs/kg or Roentgen units? (The coulomb is the S.I. unit of charge, defined as that charge transported per second by an electrical current of 1 ampere).

If each incoming photon produced C coulombs/kg, then the exposure X produced by N photons would be N × C, and a counts per minute scale could be recalibrated to read directly in exposure rate (*e.g. C/kg·h or mR/h*).

However, the ionization produced by a photon is not constant, but depends to some extent on the energy of the photon. Fortunately, the variation with energy is not great, and it can be further reduced by installing specially designed filters or attenuators round the G.M. tube ("energy-compensated tube"). Thus a typical variation might be ± 20% for energies between 0.1 and 1.5 MeV, which is considered acceptable for radiation safety purposes.



G.M. Calibration Curve

The C/kg·h or (mR/h) scale on such a G.M. exposure meter is often determined for the energy of a ^{60}Co (1.25 MeV) or ^{137}Cs (0.66 MeV) source, and a deviation curve showing the error at other energies supplied with the instrument.

Some modern instruments are calibrated in units called "grays", which are units of "absorbed dose", or "kerma", and not exposure. The reasons for this are explained in the next chapter.

Miniature versions of the G.M. exposure meter are available for use as personnel monitors, replacing the previously described pocket ion chamber in some instances. While more expensive, these dosimeters have an easily read digital display, emit an audible "beep" as a constant indication of exposure rate, and some models can be programmed to sound a continuous warning at a preset accumulated exposure or exposure rate. These features perhaps make them more suitable for infrequent use by relatively untrained staff, such as those in hospital emergency departments.

Personal Dosimeter

A3.7 SEMICONDUCTOR DETECTORS

Semiconductor detectors are relative newcomers, and may be viewed as "solid state" ion chambers. As the transistor evolved from the electron tube, so might it be said that semiconductor detectors evolved from gas detectors. Like the transistor, semiconductor detectors operate on the basis of "electron-hole pairs" rather than "ion pairs", but may be treated much the same for our purposes.

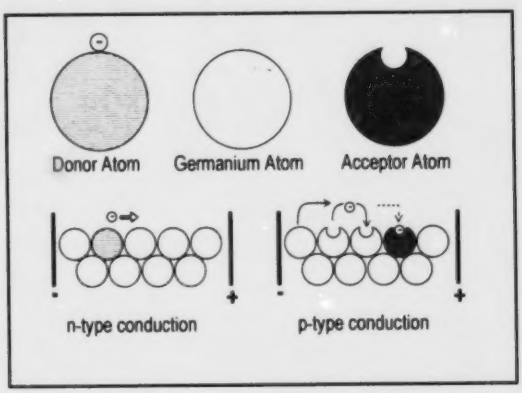
The first immediate advantage of substituting a solid material for a gas is that there is a higher density of atoms with which the radiation can interact, thus giving a higher detection efficiency.

Solid materials may be classified into electrical insulators or conductors. If an electrical field is applied across a conductor, an electrical current results, regardless of any radiation. An insulator, on the other hand, will withstand the applied field, but electron-hole pairs generated by radiation are trapped within it and cannot be collected to produce a signal. For a solid-state ion chamber we need a material which behaves partly as a conductor and partly as an insulator, i.e. a semiconductor.

There are numerous compounds which are semiconductors, and many are being explored as potential radiation detectors, but the two most common and well-known materials are silicon and

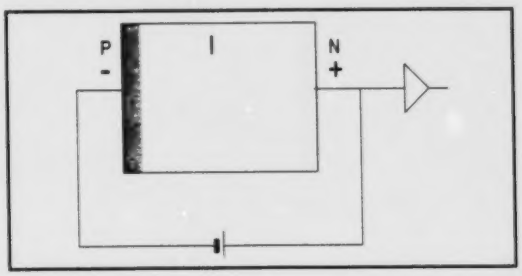
germanium. Silicon detectors have many advantages, especially for detecting charged particles and low energy photons (x and gamma), but germanium, because of its higher atomic number, is preferred for most gamma ray detection.

Electrical conduction in germanium and other semiconductors actually depends on the deliberate introduction of foreign or impurity atoms, a process referred to as "doping". If the dopant atom has an extra electron outside a completely filled orbit, the extra electron is easily detached from the atom. Such a dopant atom is called a "donor" because it donates electrons for conduction. A semiconductor doped with such atoms is called an "n-type" semiconductor, because the conduction is via negative charge carriers. If the dopant atom is missing one electron from an otherwise filled shell, it is called an "acceptor" or "p-type" dopant.



When an electric field is applied to n-type germanium, the loosely attached electrons of the donor atoms pull free and carry the current. This is normal electrical conduction, as for example, in a metal. In p-type material, the vacancy or hole of an acceptor atom combines with the electric field to attract an electron from a nearby germanium atom. The electron detaches from the atom, leaving a hole in its place, and fills the acceptor vacancy. The new hole attracts another "upstream" germanium electron to fill it in turn, and so on, so that electrons hop from atom to atom towards the positive side of the electric field. Viewed differently, the current is carried by the hole moving from right to left in the figure above, towards the negative electrode.

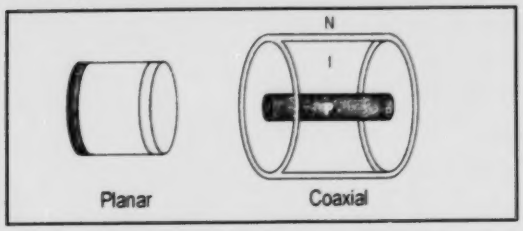
Undoped germanium is neither n or p type, but has the normal, intrinsic properties of the element, and is therefore called "intrinsic". The radiation-sensitive volume of a semiconductor detector is actually composed of such intrinsic material, sandwiched between regions of n and p type material, to which electrical contacts are made.



The P-I-N construction acts as a diode. When an electric field is applied as shown above, the diode is "reverse biased", and very little current flows. Radiation entering the intrinsic region creates electron-hole pairs which are collected by the applied field to produce a pulse of current which is electronically amplified.

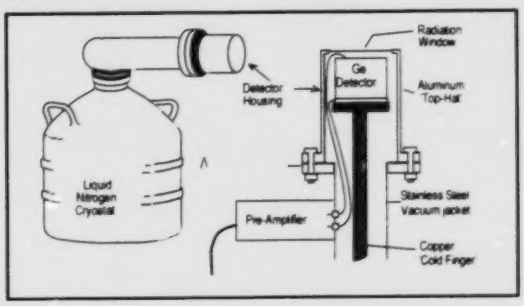
To be intrinsic, the germanium must be extremely pure. In the earlier years of semiconductor detector development, purification techniques could not reach the required purity, and the lithium drift process was used to produce intrinsic germanium. This consists of coating one face of a p-type germanium crystal with lithium, and applying an electric field under controlled conditions. The lithium, which is an n-type impurity, "drifts" through the crystal and neutralizes or compensates the p-type impurity atoms already present in the material. A detector produced by this process is called a Germanium Lithium-Drifted or Ge(Li) detector.

The figure below shows a planar configuration where the lithium was drifted from one face of a cylindrical crystal towards the other face, and a coaxial configuration, where the drift was from the outer cylindrical surface, in towards the centre.



It has since become possible to purify germanium to the required standard, and hyperpure Germanium or HpGe detectors are the result.

Besides radiation, heat also produces charge carriers in germanium. Even at ordinary room temperature, such thermally generated noise is unacceptable, and germanium detectors are therefore cooled with liquid nitrogen by mounting them in a vacuum cryostat as shown in the figure below.



The size of the output pulse from such a detector is very precisely proportional to the energy deposited by a gamma ray, making the device a high resolution Gamma ray spectrometer, and the instrument of choice for identifying radionuclides from their gamma ray spectra. (Gamma ray spectroscopy is discussed in section A3.9)

Personal monitors based on semiconductor detectors are now available commercially (eg. Siemens-Plessey) but still relatively expensive.

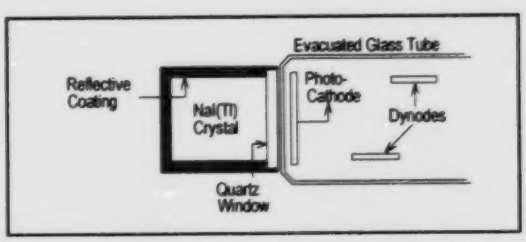
A3.8 SCINTILLATION DETECTORS

When an ionizing particle interacts with an atomic electron, it may fail to completely detach the electron from its atom (ionization), but merely transfer sufficient energy to raise the electron to a higher energy state (excitation). When the electron subsequently falls to a lower level, the excess energy is emitted as electromagnetic radiation, often in the visible energy range. This is the basis of the scintillation detector, the luminous radium watch dial, and similar "glows" often associated with radioactivity.

A3.8.1 NaI(Tl) DETECTOR

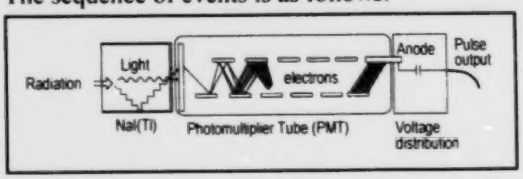
Pure sodium iodide is not a scintillator at room temperature, but when "activated" or doped with a small amount of thallium it becomes one of the most commonly used gamma ray detectors. When thallium atoms are incorporated into the structure of a sodium iodide crystal they produce impurity energy levels. Electrons are more easily excited to these levels, and on de-excitation, produce visible light of a wavelength not readily absorbed by the bulk sodium iodide.

The NaI crystal is sealed in a cylindrical aluminum container to prevent it absorbing atmospheric moisture and losing its ability to transmit scintillation photons to the photocathode. One face of the container is thin to permit radiation entry, and for low energy gamma or x-ray energies (10 to 50 keV), this window may be made of beryllium or Mylar. The opposite face of the container consists of a quartz window which is transparent to the near ultraviolet scintillations produced, and the inside surface of the container is coated with a material (aluminum oxide) which reflects this wavelength of light.



The quartz window is optically coupled by a light-transmitting gel to the face of an evacuated electron tube of special design, called a photomultiplier tube (PMT), which is plugged into a base designed to supply voltage to the various electrodes, and collect the signal pulse from the PMT anode.

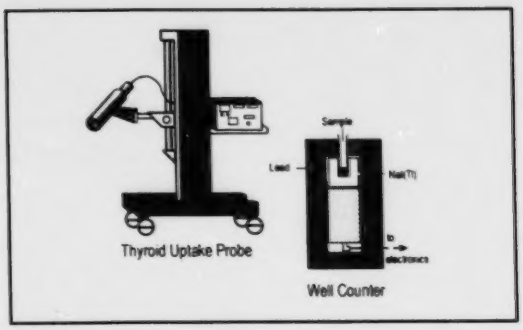
The sequence of events is as follows:



1. the ionizing photon enters the crystal and imparts some (Compton) or all (photoelectric) of its energy to an atomic electron
2. this electron dissipates its energy through the creation of excitations and other processes
3. deexcitations occur, emitting an amount of scintillating light which is proportional to the energy dissipated
4. the light strikes the photocathode of the PMT, and produces electrons
5. the electrons emitted by the photocathode are accelerated to the first dynode by the voltage applied to the tube, and release a greater number of electrons on striking the dynode. These electrons are in turn accelerated to the second dynode, and so on down the PMT, until the anode collects an amplified signal pulse whose size is proportional to the energy deposited in the Nal(Tl) crystal.

Before the advent of the germanium detector, Nal(Tl) detectors were widely used for gamma ray energy measurements, and because of their higher detection efficiency and lower cost, they are still common for such applications where energy resolution (see section A3.9) is not paramount.

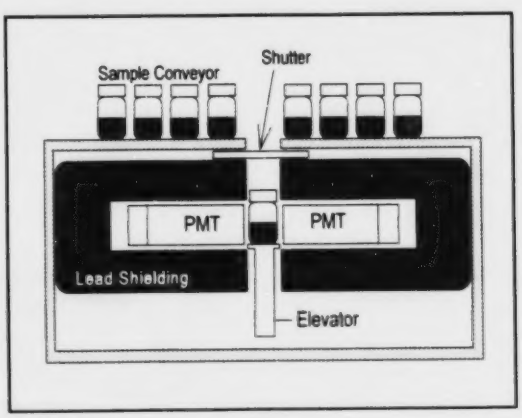
Typical applications include monitoring for internal human contamination (as in the Thyroid Uptake Probe and Whole Body Counter), well counters for the analysis of small samples, and the scintillation or gamma camera used for acquiring nuclear medicine images.



A3.8.2 LIQUID SCINTILLATION DETECTORS

Some radionuclides (e.g. ^{3}H, ^{14}C) emit only low energy beta particles, or low energy x or gamma rays (e.g. ^{125}I) which may be severely or totally attenuated by the window material of any previously described detector. Such radionuclides are best measured by mixing them with a liquid scintillator which absorbs the radiation energy and emits light, which is detected and amplified by photomultiplier tubes as previously described.

The liquid scintillator is a "cocktail" of organic solvent (e.g. toluene), one or more fluorescent solutes, and sometimes other agents to assist the dissolution of the radioactive sample. A commonly used automatic sample counter employing liquid scintillation is depicted below.



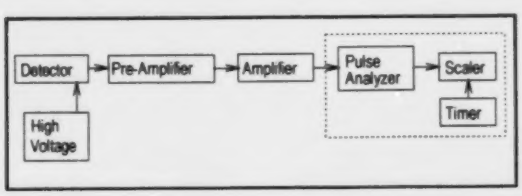
A conveyor belt carries up to several hundred pre-mixed samples for counting. Each sample vial is lowered in turn to the counting position between a pair of photomultiplier tubes for a preset counting time, often controlled by a microprocessor. The output pulses of the PMT's are summed to provide a measure of the total scintillation light output, proportional to the energy deposited in the liquid scintillator. Scintillations produced simultaneously by a single ionizing particle will register simultaneously in both phototubes, and by counting only such coincident pulses, random background and electronic (dark current) noise is almost eliminated.

The actual scintillator "cocktail" used, the mixing, preparation and standardization of samples, depends on the specific application, and is beyond the scope of this simple description. This process can be now simplified with the purchase of filter papers impregnated with liquid scintillator.

In radiation protection, liquid scintillation is most often used for measuring the activity of biological samples (e.g. urine) to assess the amount of beta-emitting radionuclides ingested by the subject.

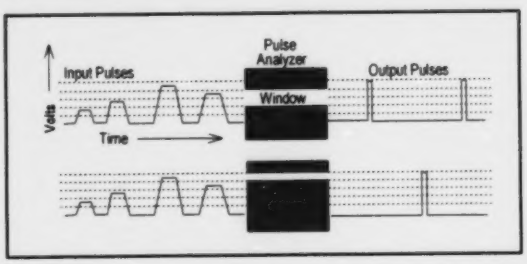
A3.9 ENERGY SPECTROSCOPY

As we have seen, some radiation detectors generate an output pulse whose size is proportional to the energy deposited in the detector by the ionizing radiation. Important examples of these are the germanium and NaI(Tl) detectors, used to determine the energy spectrum of a Gamma ray emitter. The electronic components of the spectrometer system perform similar functions for both types of detector, though there are differences in the actual equipment used. A block diagram of such a system is shown below.



In more sophisticated (and expensive) systems, the components enclosed in the box would be replaced by a single instrument called a multichannel analyser (MCA).

The high voltage unit supplies about 1000V for the PMT of a scintillation detector, and about 3000V for a germanium detector. The pre-amplifier acts as an impedance matcher. In the germanium detector, the preamp has a low noise charge-sensitive design, often with the first stage field effect transistor cooled by mounting it inside the cryostat. The main amplifier shapes and amplifies the pulses. In simple, economically priced systems, the analogue pulses (pulses whose size varies according to the energy deposited) are fed into a single channel analyser which has an electronic "window" whose level and width can be adjusted by hand.



If, for example, the level is set at 2V and the window width at 2V, then only those incoming pulses with sizes between 2V and 4V will generate an output pulse. These output pulses are "digital", i.e. of constant amplitude regardless of the size of the corresponding input pulse, and are fed to a pulse-counting device, commonly called a "scaler" for historic reasons. The counting period of the scaler is controlled by a timer.

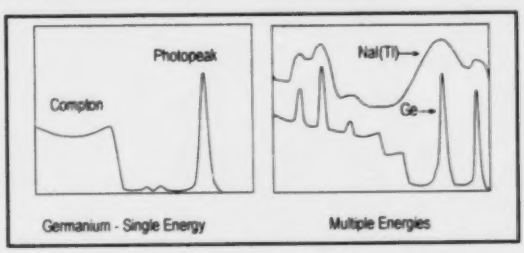
By setting the analyser level at, say, 1V, and the window width at 0.1V, all incoming pulses with amplitudes between 1 an 1.1V may be counted for a fixed period (say 10 seconds). Keeping the window width fixed, the level may be raised to 1.1V and pulses between 1.1 and 1.2V counted for the same 10 second period, and so on. In this way, a histogram or spectrum may be plotted relating

pulse height to the number of pulses having that height. The multichannel analyser automatically sorts the incoming pulses according to size and stores and displays the spectrum.

Since the height of any incoming pulse is proportional to the energy deposited by the gamma ray that caused it, and the number of pulses per unit time is a measure of radiation intensity, this is also an energy spectrum.

Note that the energy registered is the energy imparted to an **electron** when the gamma ray interacts in the crystal via the Compton or photoelectric effects. Recall from Chapter A2.3 that an electron produced by the photoelectric effect has essentially all the energy of the original gamma ray, and such electrons produce the "full energy absorption peak" or "photopeak" at the upper end of the spectrum. Compton electrons carry varying amounts of the original gamma energy, depending on the interaction angle, and register as a broad continuum or "Compton background".

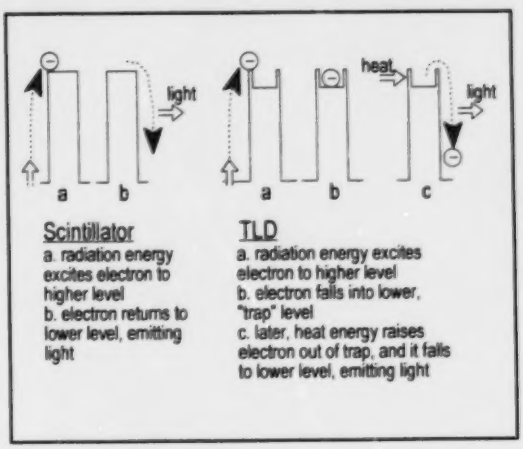
The figure below shows a germanium detector spectrum of a monoenergetic source, and of a source emitting multiple gamma energies, which demonstrate the advantages of the germanium spectrometer's superior energy resolution.



Computer programs are available to automatically analyze such spectra, based on the areas and energies of the photopeaks, and after calibration, to identify the various radionuclides in a source sample, and quantitate their activities.

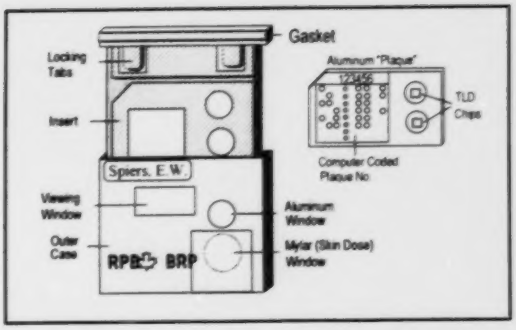
A3.10 THERMOLUMINESCENT DOSIMETRY (TLD)

We have seen how, in scintillation detectors, incident radiation energy raises electrons to higher or excited energy states, which quickly deexcite and give up their energy in the form of light scintillations. In thermo-luminescent materials, such as lithium fluoride (LiF), calcium sulphate ($CaSO_4$), etc. electrons excited to certain energy levels are trapped there more or less indefinitely, but may be induced to deexcite at a later time by heating, hence the name thermo- (heat) - luminescent (light emitting). A schematic diagram of this process is shown in the figure below.



The amount of light released by heating depends on the number of electrons trapped in excited levels, which is in turn proportional to the amount of radiation the TLD material has absorbed, so the system can be calibrated to give radiation dose as a function of light output.

This is the basis of the National Dosimetry Service operated by the Radiation Protection Bureau of Health Canada, and is used to monitor and record the radiation doses of the vast majority of Canadian radiation workers. A TLD badge is shown on the next page.



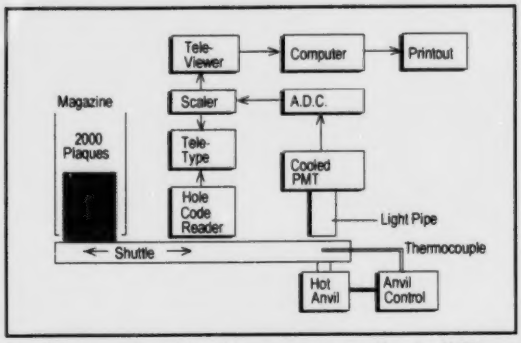
Two "chips" of TLD material (lithium fluoride) are mounted on an aluminum carrier or "plaque". The plaque has a serial number printed on it, and the same serial number is punched out as holes in the metal (BCD computer-readable code).

The plaque fits (one way only) into a plastic tray having circular cut-outs to permit unimpeded irradiation of the TLD chips, and the tray slides into an outer plastic case and locks in position. A transparent window allows a view of the plaque number when the badge is sealed.

There are two sets of radiation windows, front and back, corresponding to the locations of the two TLD chips. The lower window covers consist of a thin film of aluminized Mylar with attenuation equivalent to that of human skin (7 mg cm^{-2}), so that the chip behind it is exposed to and measures the badge wearer's "skin dose". The other chip resides behind 2 mm of aluminum, and therefore registers only the more penetrating radiations, as a measure of depth or "total body dose".

Each quarter, the RSO collects the badges from individual wearers, opens the cases with a special tool, replaces the old TLD plaques with new ones, and sends the old plaques to Radiation Protection Bureau to be read. A computerized printout of results is returned to the RSO.

At Radiation Protection Bureau in Ottawa, the plaques are loaded into the hopper of an automated TLD readout system shown below.



In lithium fluoride there are actually 5 different thermoluminescent "trap" levels. At room temperature electrons remained trapped in these levels for half lives of about 5 minutes, 10 hours, 0.5 years, and 7 and 80 years respectively. Since there is significant decay from the first 3 levels during the 3 month wearing period and the subsequent handling period before readout, these levels are not used for dosimetry. Before reading the dosimeter, these levels are "emptied" of any residual electrons by controlled heating or "annealing" in a separate oven for 1 hour at 80°C. The remaining two "deeper" (and hence longer lived) levels on which the dosimetry is based, require a temperature of 270°C for deexcitation.

Each plaque is carried in turn by the shuttle to the optical code reader where the plaque identification number is read, printed on the Teletype, and transmitted to the computer. The plaque then continues to the readout head, where the thicker of the two chips is positioned under the light pipe. The hot anvil, at 270°C, rises to meet the lower surface of the chip, pressing the upper surface against the thermocouple probe. When the thermocouple indicates that the upper surface has reached 270°C (about 8 seconds), the anvil drops, and the process is repeated on the thinner chip (about 4 seconds). During the heating cycle, the emitted thermoluminescent light is carried along the light pipe to the photomultiplier which converts it to a proportional electrical signal (see section A3.8.1). The PMT output is converted to a string of digital pulses by the Analog-to-Digital Convertor (ADC), and the pulses are counted by the scaler to give a number which is indicative of

the light output, and therefore of the radiation absorbed by the dosimeter. After every 200 plaques, a standard is inserted, which consists of a plaque that has been exposed to a known dose of radiation, and the calibration is rechecked.

The system is highly linear and reliable for radiation doses between 0.2 mGy - 10 Gy (20 mrad-1000 rad). Since doses below 0.2 mGy are not reliably measurable, they are not reported in the Radiation Protection Bureau system. Also, low energy Beta particles, such as those emitted by ^{3}H for example, will not penetrate even the thin mylar window of the dosimeter, and are therefore not detected by it. Betas from ^{14}C are sufficiently energetic to just penetrate the window, but activities of ^{14}C high enough to generate a significant reading are rarely used in radionuclide laboratories.

As well as keeping track of names and plaque numbers for dosimeters undergoing readout, the computer system also stores records of each individual's accumulated dose to date in a database called "The National Dose Registry". In addition to the lapel badge described above, extremity dosimeters in ring or bracelet form are available.

Further information on the administrative aspects of the Radiation Protection Bureau Dosimetry Service are contained in Chapter B6. More detailed information on the dosimeters and the automated TLD readout system is contained in reference RP78 (page *A3-16*).

A3.11 Quantitative Factors

Assessing the presence and amount of radioactivity is invariably accomplished by detecting the radiation emitted. The difference between the amount of radiation emitted and the amount actually detected depends on a number of factors:

1. Counting Geometry
2. Detector Efficiency
3. Deadtime
4. Background Radiation
5. Counting Statistics.

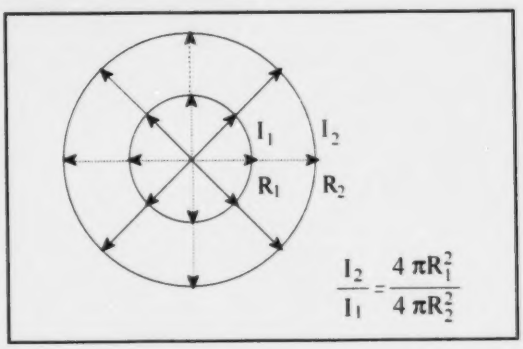
A3.11.1 COUNTING GEOMETRY

Those factors which determine whether the radiation hits or misses the detector are collectively referred to as "counting geometry". These factors are:

1. the geometric relationship between the source of the radiation and the detector, and
2. the presence of material which absorbs or scatters the emitted radiation.

Charged particles, such as Beta particles, are readily absorbed or scattered by air, so that their accurate measurement requires that the detector be close to or in contact with the source. X-rays and gamma rays are less affected by air and can be detected at some distance from the source. Under these circumstances the **inverse square law** can be applied, which states that the intensity of the radiation varies as the inverse of the square of the distance from the source.

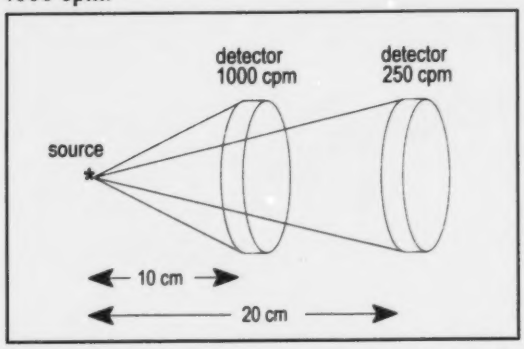
To understand why this is so, consider two concentric spheres of radii R_1 and R_2 with a gamma source at their common centre.



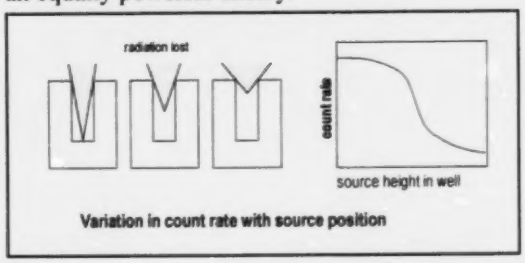
Assuming that the radiation is emitted isotropically, and that there is no attenuation, then the same radiation (say N photons per second) will pass uniformly through the surfaces of both spheres. But radiation intensity is expressed as the number of photons per second passing through unit area. So, to find the intensity at the surface of each sphere we must divide the total radiation by the surface area of the sphere. Thus the intensity at the

surface of the first sphere is N divided by $4\pi R_1^2$, and N divided by $4\pi R_2^2$ at the surface of the second. The ratio of intensities is therefore $I_2/I_1 = R_1^2/R_2^2$.

Thus, if a radiation detector is positioned 10 cm from a gamma source and records a count rate of 1000 counts per minute (cpm), then doubling the distance to 20 cm should reduce the count rate by a factor of four, i.e. to 250 cpm. ($I_2/1000 = 10^2/20^2$, so $I_2 = 1000 \times 100/400 = 250$). Conversely decreasing the distance by a factor of two to 5 cm will increase the intensity by a factor of four to 4000 cpm.



If we substitute a person for the detector in the above example we can see that the inverse square law can be a powerful ally in any attempt to reduce radiation exposure. On the other hand, if we have a case of skin contamination, for example, it can be an equally powerful enemy.



A radioactive source emits radiation in all directions, i.e. into 4π steradian of solid angle. If we wished to detect all emitted radiations, we would have to have the detector completely surround the source. The well counter is a convenient approximation to this situation.

Note from the diagram that some of the emitted radiation is lost through the top hole of the well, and that more radiation is lost if the source extends to the top.

Material in the vicinity of the source or detector will alter the amount and energy of radiation reaching the detector due to photoelectric absorption or Compton scatter. Such material includes the material of the source itself, for gamma rays emitted by an atom deep inside the source may be absorbed or scattered before they leave the source. One might therefore, for example, expect to measure more photons from an activity of 1 MBq of I-125 dissolved in 1 mL of water than the same activity dissolved in 10 mL of water. This effect is known as "source self-absorption".

Material near the detector may actually increase the count rate by scattering (reflecting) radiation back into the detector that might otherwise have missed it.

It should now be obvious that when two sources are to be compared for activity, as in the use of a calibration source for example, the counting geometries should be as identical as possible for both sources.

A3.11.2 DETECTION EFFICIENCY

Simply put, detection efficiency is the percentage of the total radiation hitting the detector that is actually detected. The majority of gamma rays, for example can quite easily pass through a gaseous ion chamber without interacting with a single gas molecule. Thus, gas detectors have a low detection efficiency for gamma rays. Sodium iodide or germanium detectors, being solid, have higher detection efficiencies. Since these latter detectors are most frequently used for spectroscopy, where the photopeak is used as an indication of the intensity of a particular gamma ray energy, a "photopeak efficiency" is usually quoted, which is the number of counts registered in the photopeak as a percentage of the total number of gamma rays hitting the detector. The detection efficiency of various detector types is further discussed in chapter B5.

A3.11.3 DEAD TIME

Detector systems designed for quantitative measurements have electronic equipment to count and record the number of pulses received from the detector. All such equipment requires a finite time to register a pulse, and during this time, it is "dead" to any other pulses which may arrive. Thus, if the count rate is too high and the pulses are arriving too close together, many of them will be lost. The higher the counting rate, the higher the fraction of pulses lost.

Counting electronics can be characterised as "paralyzable" or "non-paralyzable". In a paralyzable system the arrival of a second pulse during a dead time merely extends the dead time. If the count rate is sufficiently high, such a system can be dead all the time, i.e. paralysed, with an apparent count rate of zero. This situation can be potentially hazardous if it causes a survey meter to read zero in a high intensity radiation field.

Under more normal circumstances, the easiest way to avoid counting errors due to dead time losses is to keep the count rate reasonably low. Some equipment, such as multichannel analysers, often monitor and compensate for dead time by automatically extending the time of acquisition (the "livetime") beyond the preset counting time.

Reference NC85 (page *A3-16*) contains a comprehensive discussion of dead time.

A3.11.4 BACKGROUND RADIATION

Cosmic radiation, natural radioactivity in the earth and building materials, as well as sources of unnatural radiation such as wandering injected patients, all constitute unwanted "background radiation", which can interfere with a measurement.

The two common methods of dealing with this are:
1. shield the detector, and
2. measure the background separately with the source removed and subtract it from the results of the measurement.

A3.11.5 COUNTING STATISTICS

Radioactive decay is a random process: we can never know exactly when a particular unstable nucleus will decay, but can only deal in the average behaviour of billions of such unstable nuclei. As a result, the counts per minute from a radiation detector are subject to statistical variation. If we placed a long lived source in a well counter and performed 15 repeated 1 minute counts, we might get the following numbers of counts: 99, 97, 102, 80, 104, 112, 98, 119, 89, 111, 101, 92, 108, 91, 99.

The mean or average count rate is 100 cpm, but any single count might yield a quite different result. The more counts we do, or the longer the time we count, the more likely it is that our cpm result will be close to the "true" result.

The results actually follow what is known as a "Poisson distribution", a statistical theory which applies to situations where the probability of an event is low, but a significant number of events occur anyway because they are drawn from a large population. In our case, there are billions of unstable nuclei with the potential for transformation, but relatively few of them undergo the transformation in any given time interval. (Poisson originally tested his theory by perusing the records of Prussian cavalry regiments to determine how many soldiers had died each year from being kicked by their horses).

From the Poisson theory, it can be shown that if we have a result N counts per minute, then a measure of the uncertainty in this result, called the "variance of the mean", is also N. Another measure of uncertainty in the result, the "standard deviation" (S.D.) is given by $\sqrt{N}$.

Thus, if we obtained a result of 100 counts, the variance is 100 and the S.D. is $\sqrt{100} = 10$. In this case the standard deviation is 10% of the number of counts (10/100 = 10%).

Suppose we recorded 2700 counts from a sample in a time of 3 minutes, removed the sample from the counter, and recorded a background of 300 counts in another 3 minutes. The sample count rate is 2700/3 = 900 cpm and the background is 300/3 = 100 cpm. To find the "true" count rate we subtract the background from the sample:
i.e. 900 - 100 = 800 cpm.

It can be shown that if N counts are collected in time t, then the standard deviation on the count rate (N/t) is given by $\sigma = \sqrt{N / t^{2}}$.

Thus, the standard deviation on the sample count in the above example is $\sigma_s = \sqrt{900/9}$, and the standard deviation on the background count is $\sigma_b = \sqrt{300/9}$. It can also be shown that the standard deviation on the difference (or the sum) of these two results is $\sigma = \sqrt{\sigma_s^2 + \sigma_b^2}$. So the standard deviation on the 800 cpm net count rate is $\sqrt{2700/9 + 300/9} = 18.3$ cpm. The percent standard deviation is then (18.3/800) × 100% = 2.3%

References

RP78 The Thermoluminescent Dosimetry Service of The Radiation Protection Bureau, Health and Welfare Canada, 79-EHD-27, December, 1978. Obtainable from Information Directorate, Health Canada, Brooke Claxton Building, Ottawa, K1A 0K9.

NC85 A Handbook of Radioactivity Measurements Procedures, NCRP Report No. 58, 2nd Edition, February 1985, The National Council on Radiation Protection and Measurements, 7910 Woodmont Avenue, Bethesda, MD 20814.

CHAPTER A4

RADIATION DOSIMETRY UNITS

A4.1 INTRODUCTION

Over the past fifty years or so, a system of concepts, definitions and units has evolved for quantifying ionizing radiation and its biological effects. The system is still evolving, with new definitions being added and more precise definitions replacing older ones almost annually. The International Commission on Radiation Measurements and Units (ICRU) regularly publishes reports with the latest information in this area. (see chapter A7 for more information on the ICRU)

Canada has now fully adopted the International System of units (S.I.), and these should be used in all practical situations, where possible.

However, it is felt that a Radiation Safety Officer should still have some knowledge of the older units and their conversions to S.I. units, since he or she is likely to encounter them in older publications, or even in older pieces of equipment still in active service.

In this chapter we will discuss the following.

Activity: The "strength" of a radioactive source, in terms of the number of nuclear transformations occurring per second. The S.I. Unit is the becquerel (Bq), and the old unit is the curie (Ci).

Exposure: A measure of the amount of ionization produced in air by photons (x and gamma rays). The S.I. unit is the coulomb per kilogram (C/kg) and the old unit is the röntgen (R).

Exposure rate: The S.I. unit is the coulomb/kg·h (C/kg·h), and the old unit is the röntgen/h (R/h).

Exposure rate constant: The exposure rate 1 meter from a 1 becquerel source of a specified radionuclide. The S.I. unit is the $C/kg \cdot h \cdot Bq \cdot m^2$, and the old unit is the $R/h \cdot Ci \cdot m^2$.

Absorbed Dose: The amount of radiation energy absorbed per unit mass of a target material. The S.I. unit is the gray (Gy) and the old unit is the rad.

Air Kerma: Kerma to air is often used to replace the old unit of exposure. For medical radiations, exposure in R can be converted to air kerma in mGy by multiplying by 8.73.

Equivalent dose: Absorbed dose modified to take account of the increased biological damage produced by some types of radiation. The S.I. unit is the sievert (Sv), and the old unit is the rem. Equivalent dose used to be called dose equivalent in ICRP 26 (reference IC77, page *A4-6*).

Effective dose: Equivalent dose modified to take account of the fact that partial body dose is less detrimental than whole body dose, and that some human organs are more sensitive to radiation than others. The S.I. unit is the sievert, and the old unit is the rem. Effective dose used to be called effective dose equivalent in ICRP 26.

Committed dose: An estimation of the radiation dose a person is committed to receive over a future time period because they have become "internally contaminated" by an intake of radioactive material.

A4.2 ACTIVITY

We have already discussed activity in chapter A1, but it is included here again for the sake of completeness.

A radioactive source is said to have an activity of one becquerel (Bq) if one nuclear transformation or "disintegration" occurs per second in the source.

The old unit of activity, the curie (Ci), was defined as the activity of a source in which the disintegration rate was 3.7×10^{10} disintegrations per second (dps), which correspond to the activity of 1 g of radium.

The nuclear transformations produce radiation, but there is no simple correspondence between the number of transformations and the amount of radiation produced; each radionuclide emits radiation in accordance with its own decay scheme.

Because of their different physical half lives, types of radiations emitted and biological behaviour if ingested, some radionuclides are considered more hazardous ("radiotoxic") than others. Thus, for example, 1 kBq of I-131 is considered to be about as hazardous as 100 kBq of Tc-99m. This is reflected in the "scheduled quantity" figure assigned to each radionuclide and discussed in chapter B1. The radiotoxicity of a radionuclide also depends to a great extent on its chemical and physical form, and the route of entry into the body, but it is impossible to capture all these variables in a single parameter, and the scheduled quantity is intended to be a regulatory simplification.

The "specific activity" of a radioactive sample is the activity per unit mass, e.g. Bq per gram.

A4.3 EXPOSURE

One of the earliest means of detecting x-rays was by collecting and measuring the ions they produced in air (see chapter A3). It is not surprising, therefore, that one of the first units proposed for expressing quantity of radiation was based on the number of ions produced in air.

In 1908, Villard suggested a nameless unit for a quantity of x-rays as that amount which produced 1 esu of charge in 1 cm^3 of air at S.T.P. (i.e. 0° C and 760 mm pressure). In 1928, the ICRU defined this unit as the röntgen, designated by the symbol "r" (since changed to "R").

In 1938, the ICRU included gamma rays in the definition, and substituted the mass of 1 cm^3 of air at S.T.P. (0.001293 g). In 1956, they applied the term "exposure dose" to this ionization based measure of radiation, and in 1962, dropped the word "dose" to avoid confusion with the quantity "absorbed dose" (see below).

Several other modifications were made in subsequent years, including the use of the Coulomb for charge and a capital "R" for the symbol, in keeping with the S.I. system of units, so that the definition of exposure became:

"The exposure, X, is the quotient dQ by dm where the value of dQ is the absolute value of the total charge of ions of one sign produced in air when all the electrons (negatrons and positrons) liberated by photons in air of mass dm are completely stopped in air. $X = dQ/dm$."

Exposure Rate is simply the rate at which exposure is produced. A survey meter typically reads exposure rate in C/kg·h, or in mR/h if it is an older instrument. Knowing the exposure rate at a location, one can quickly estimate how long a person can remain there before their total, accumulated exposure exceeds a certain value.

Some newer instruments are actually calibrated to read in grays per hour (Gy/h). The gray is a unit of absorbed dose (see below), not exposure. The reasons for this confusion are discussed in section A 4.11 at the end of this chapter.

A4.4 Exposure Rate Constant

When exposure is due to gamma or x-rays from a radioactive source (as opposed to an x-ray machine, for example), it is possible to relate the exposure rate to the activity of the source.

In 1951, the ICRU suggested that gamma ray emissions from a radionuclide could be expressed in terms of R per mCi·h at 1 cm, and this became known as the **"k factor"** for a radionuclide. The name "specific gamma ray emission" was introduced in 1957, and **"specific gamma ray constant"** in 1962.

Until this point, only gamma ray emissions were included in the definition. But radionuclides may also emit x-rays and internal bremsstrahlung, so in 1971 the name was changed to **"exposure rate constant"**, and the definition reworded to include all photon emissions above a certain threshold energy.

In 1980, the concept was again modified: since the ultimate purpose is to determine the absorbed dose to something (or someone) irradiated by the radionuclide emissions, and since absorbed dose is expressed in gray (see below), air kerma was substituted for exposure, and the constant renamed the **"air kerma rate constant"**.

Thus, air kerma rate constant, Γ_δ, for a particular radionuclide is the number of grays produced per second in air by photons with energy greater than δ, at a point 1 m from a source of 1 Bq.

The concept of kerma has not yet been fully embraced by the general radiation protection community, so that exposure rate constant or even specific gamma ray constant may be more commonly used at present.

Note that exposure, measured in röntgens, may be converted to air-kerma, measured in milligray, by multiplying by the conversion factor 8.73. (see example 2.4.3 in Appendix XA2 for the derivation) providing bremsstrahlung production can be neglected (i.e. for most radionuclides encountered in a hospital setting).

Knowing the exposure rate constant for a particular radionuclide, and assuming that the radiation obeys the inverse square law, we can calculate the exposure rate at any distance from a source of known activity.

We should note that all of the above presupposes an ideal point source, which is never realized in practice, and neglects the scattering effects of air or any other medium between the source and point of measurement.

A4.5 Absorbed Dose

The effect of radiation on a biological system is directly proportional to the amount of energy absorbed. A first important step, therefore, in predicting biological effects, is an assessment of the amount of radiation energy absorbed.

The first suggestions for a dosimetry unit based on energy absorption appeared in the early 1900's. The idea was given further impetus in the late 1930's, following the discovery of the neutron and the subsequent requirement for some means of measuring neutron radiation. Because of the different interaction mechanisms, the exposure concept, in use for x (and gamma) rays, was not applicable.

In 1951, the ICRP recommended that for correlating the amount of any radiation with its biological effects, the "dose" be expressed in energy absorbed per unit mass (ergs per gram of absorber). In 1954, they gave this quantity the name "absorbed dose" and defined the rad (radiation absorbed dose) unit as 100 ergs per gram.

Many refinements were made to the formal definition of the rad in subsequent years until, with the conversion to S.I. in 1975, a unit of 1 joule per kilogram was adopted, and given the special name "gray" (Gy), after Louis Gray, a pioneer of radiation dosimetry.

Since 1 joule = 100,000 ergs, and 1 kg = 1000 g, it follows that
1 Gy = 1 J/kg = 100 ergs/g = 100 rads.

A4.6 Equivalent Dose

In chapter A2 we saw that some radiations, such as alpha particles, laid down energy more densely along their tracks than did electrons. Such densely ionizing, or high LET radiations produce more pronounced biological effects than the low LET electrons, for the same amount of energy deposited per unit mass (i.e. for the same absorbed dose).

In radiobiological experiments, a factor known as "the relative biological effectiveness" (RBE) has been used to compare the biological effect of one type of radiation to another. The standard radiation used for comparison is usually 200 kV_p x-rays (radiation from an x-ray machine with an operating voltage of 200 kilovolts).

In 1962, with the introduction of the rad unit of absorbed dose, the ICRP also introduced the quantity "dose equivalent", H, to replace the old RBE dose. In 1977, they defined dose equivalent in rems as H = D Q N where D is the absorbed dose in rads, Q is a numerical factor called the "quality factor", and N is a number representing all other factors which might modify the harmful biological effects of the radiation.

Fortunately, few radionuclides in present use in hospitals emit radiations other than x-rays, gamma rays and beta rays, all of which have a recommended Q value of 1.

With the introduction of the gray as the S.I. unit of absorbed dose, a new unit for dose equivalent was also required, and was given the name "sievert", after the Swedish health physicist, Rolf Sievert. Just as 1 Gy = 100 rads, so 1 Sv = 100 rem.

Different radiations are known to have a different biological effect for the same given absorbed dose. Over the years a number of different methods have been used to account for this. As the reader may often find these quantities mentioned in publications they are briefly dealt with here.

With the publication of Report #60 by ICRP (reference IC91, page *A4-6*) there has been some simplification of the suggested radiobiological quantities. ICRP have introduced the **equivalent dose H_T (Sv)** for any tissue which is calculated by multiplying the average absorbed dose for any tissue by the radiation weighting factor w_R. For the radiations used in medicine w_R is almost always 1. For gamma and x-rays the equivalent dose is numerically equal to the quantity **dose equivalent** previously used by ICRP in 1977.

A4.7 Effective Dose

To account for the different cancer risk for different tissues ICRP have introduced the **effective dose E (Sv).** This is the weighted sum of the doses to individual organs or tissues.

$$E = \Sigma W_T H_T$$

The tissue weighting factor W_T is related to the risk of carcinogenesis in the tissue T. Effective dose can be envisioned as the uniform equivalent dose which would have to be given to the whole body for the same numerical risk. There are only slight differences between the effective dose and the **effective dose equivalent** which was the previous quantity used by ICRP.

There is some controversy over the use of effective dose as a description of detriment during medical procedures.

A4.8 COMMITTED DOSE

Radioactive material accidentally taken into the body will undergo some kind of biological distribution before being excreted or permanently incorporated. At present, not much can be done to influence the natural course of events in such a situation, and someone suffering such an intake is therefore "committed" to receiving a radiation dose from their internal contamination.

The biological behaviour of many elements and compounds has been estimated by experimentation and use of mathematical models, so that the future distribution of an intaken material can be determined at the time of the intake, or shortly thereafter, and the future radiation dose profile calculated.

The equivalent dose to an organ calculated for a period of 50 years following a single intake is known as the "committed dose" to the organ due to that intake. In mathematical form,

$$H_{50} = \int \dot{H}(t)\, dt$$

where H(t) is the equivalent dose rate to organ on tissue T.

If the contributions from all organs are summed to give the effective dose received over the 50 years, this is the "committed effective dose".

We should perhaps note at this point that the names of some of the quantities we have defined are quite long, and it is not surprising that terms such as "effective dose" are often shortened to just "dose". In fact, the term "dose" is often loosely used in radiation protection for a variety of quantities.

A4.9 ANNUAL LIMIT ON INTAKE (ALI)

The annual limit on intake defined by the ICRP is that intake of radioactive material which will result in an effective committed dose to "Reference Man" of 20 mSv (the annual limit), or 200 mSv dose equivalent to any tissue, whichever is more restrictive.

A4.10 DERIVED QUANTITIES

Having calculated the ALI for a radionuclide (in becquerels), it is possible, using "average" physiological data, to calculate the environmental concentration of that radionuclide which would result in an intake equal to 1 ALI. For example, using a "Reference Man" breathing rate of 1.2 m^3/h, and assuming a work year of 2000 hours, we can calculate what concentration of a radionuclide in that air would result in an intake of 1 ALI. This is the Derived Air Concentration (DAC) for that radionuclide.

Another derived quantity is the Derived Investigation Level (DIL). When a program of monitoring for internal contamination is in place (thyroid "bioassays" for I-131, for example), it is usual to determine ahead of time, levels of contamination above which certain actions are initiated. The level which triggers an investigation and dose assessment is the Derived Investigation Level (DIL). The ICRP have suggested procedures for calculating such derived levels from the ALI.

A4.11 PROBLEMS WITH EXPOSURE

Exposure has been an important quantity in radiation protection because it is comparatively easy to measure. A G.M. or ion chamber based radiation monitor measures the charge liberated by radiation, which can be simply converted to an exposure reading. Special, highly accurate ion chambers use air as the sensitive gas, and are capable of measuring exposure to 0.5% accuracy.

Unfortunately, it is absorbed dose, not exposure, which best predicts biological effects, and exposure readings must therefore be converted to absorbed dose figures to be of use.

An argument can be made for having field instruments measure absorbed dose to water, instead of exposure (to air), and it is quite possible that the concept of exposure will be abandoned in the not too distant future.

Meanwhile, the replacement of the old unit for exposure, the röntgen, by the S.I. units C/kg, has created a more immediate problem, and a somewhat confusing solution. It can be shown (see example 2.4.4 in Appendix XA2) that an exposure of 1 röntgen in air will produce an absorbed dose of about 1 rad if the air is replaced by tissue. Thus, a radiation area which registered 2.5 mR/h on a survey meter would be said to produce a dose rate of 2.5 mrads/h, and an accumulated dose of 5000 mrads in a 2000 hour working year. An RSO could therefore immediately judge what absorbed dose would ensue from a measured exposure rate, and whether or not dose limits were likely to be exceeded.

The same calculation with S.I. units shows that an exposure of 1 C/kg would produce an absorbed dose of 37.2 Gy in tissue, so that the approximate one to one correspondence between the units of exposure and absorbed dose has been lost. Such an equivalence still exists, however, between air kerma and absorbed dose, which is why the "exposure rate constant" became the "air kerma rate constant".

In an attempt to maintain this useful correspondence between exposure and dose, some manufacturers of modern survey meters have assumed the following equality: 1 röntgen = 1 rad = 0.01 gray. Therefore, we can take an old survey meter and reprint the figures on the scale, changing, for example, "1 R" to 1 cGy" (centigray), and "1 mR" to "10 μGy", etc.

References

IC77 International Commission on Radiological Protection. ICRP Publication 26. "Recommendations of the International Commission on Radiological Protection," Annals of the ICRP, vol.1, no.3, 1977.

IC91 International Commission on Radiological Protection. Publication 60. "1990 Recommendations of the International Commission on Radiological Protection," Annals of the ICRP, vol.21, no.1-3, 1991.

CHAPTER A5

CHEMICAL EFFECTS OF RADIATION

The interactions of radiation with matter described in Chapter A2 generate ionized and excited atoms and molecules in the irradiated material. These are generally very short-lived and within about 10^{-13} second either dissociate to free radicals, react with neighbouring ions or molecules, or lose their excitation energy in the form of heat. The exact nature of the processes taking place depends on the type of material. For example, in graphite and metals almost all the absorbed radiation energy appears as heat, while with water and organic materials most of the absorbed energy is used in breaking chemical bonds to give free radicals and new chemical products.

The free radicals produced are also short-lived and within a few seconds, or less, react amongst themselves, or with the substrate, to produce chemical changes that lead to the biological effects observed much later. Radiation chemists often break the processes down into several stages, as shown in the adjoining box, each of which is some orders of magnitude longer than the preceding stage. Chemical changes lasting less than a second can bring about biological effects many years later.

Physical stage {about 10^{-18} to 10^{-13} sec}
Ions and excited molecules produced in irradiated material. Ionized and excited molecules dissociate giving free radicals and molecular products.

Chemical stage {about 10^{-13} to 1 sec}
Free radicals react to give new chemical products.

Biochemical stage {about 1 to 10^{6} sec (11 days)}
Possible impairment of biochemical functions: damage to membranes, enzymes.

Biological stage {about 10^{6} to 10^{9} sec (32 years)}
Loss of viability, sterility, cancer, genetic damage.

Direct and Indirect Action

Ionizing radiation differs from ultraviolet and visible light in not being selective in the molecules it excites. Ionizing particles and high-energy photons interact with electrons in their path without regard for the chemical form of the atom containing the electrons. This means that if a mixture is irradiated, all components of the mixture will receive some of the absorbed radiation energy but the components present in the greatest amount (i.e., providing most electrons) will receive the most energy. This gives rise to the concepts *of direct* and *indirect* action.

Direct action refers to chemical changes caused by energy deposited directly in the molecule changed. *Indirect action* occurs when the energy is absorbed by, for example, a solvent and radicals from the solvent bring about the change. Only indirect action on a substance need be considered if it is irradiated as a dilute solution, but direct action becomes more important as the concentration of the solution is increased and will be the dominant process at high concentrations and with pure substances. Since the mechanisms of direct and indirect action are different, the two processes may give different irradiation products.

Two areas that have received a good deal of attention from radiation chemists are irradiation of substances in dilute solution in water (indirect action brought about by radicals produced from the water) and irradiation of pure organic compounds (direct action). Both types of process, direct and indirect, can be precursors of biological change.

Radiolysis of Aqueous Solutions

When dilute aqueous solutions are irradiated essentially all the radiation energy is absorbed by the water. This gives rise to several reactive radicals, the hydrated electron (e^-_{aq}), the hydrogen atom (H), and the hydroxyl radical (OH), and two molecular products, hydrogen (H_2) and hydrogen peroxide (H_2O_2). Molecular hydrogen is a chemically-unreactive gas and generally escapes from the irradiated system unchanged. Compared to the radicals, hydrogen peroxide is relatively unreactive but may survive to bring about slow, post-irradiation, oxidation reactions after the radical reactions are complete.

In neutral and near-neutral solutions about 45% of the radicals formed in irradiated water are e^-_{aq}, about 45% OH, and about 10% H. However, the hydrated electrons are converted to hydrogen atoms in acid solutions (pH<3):

$$e^-_{aq} + H^+ \rightarrow H$$

The yields of radicals from irradiated water are relatively low, an absorbed dose of 10 kGy, for example, will split about 0.01% of the water molecules into free radicals (and eventually raise the temperature by about 2.5°C).

Hydroxyl radicals and hydrogen atoms react with most organic compounds by abstracting hydrogen atoms or by adding to multiple bonds. For example, with ethanol the reactions are

$$OH + CH_3CH_2OH \rightarrow H_2O + CH_3\dot{C}HOH$$
$$H + CH_3CH_2OH \rightarrow H_2 + CH_3\dot{C}HOH$$

and with ethene (ethylene)

$$OH + CH_2{=}CH_2 \rightarrow \cdot CH_2{-}CH_2OH$$
$$H + CH_2{=}CH_2 \rightarrow \cdot CH_2{-}CH_3$$

(The dot represents an unpaired electron, the characteristic feature of a free radical, but it is not always shown and is only included here as a reminder that the species are radicals. The dots are almost always omitted with simple atoms and radicals such as ·H, ·OH, and ·CH_3.)

The products of these reactions are themselves free radicals, though less reactive radicals than OH and H. They (the organic radicals) react together by combination of two radicals, for example,

$$2\ CH_3\dot{C}HOH \rightarrow CH_3CH(OH)CH(OH)CH_3$$
$$2\ \cdot CH_2{-}CH_2OH \rightarrow HOCH_2CH_2CH_2CH_2OH$$

or by disproportionation (a reaction in which a hydrogen atom is transferred from one radical to another), for example,

$$2\ CH_3\dot{C}HOH \rightarrow CH_3CH_2OH + CH_3CHO$$
$$2\ \cdot CH_2{-}CH_2OH \rightarrow CH_3CH_2OH + CH_3CHO$$

The products of disproportionation reactions always include one product with a double bond, this is the aldehyde ethanal (acetaldehyde) in the two examples shown.

Hydrogen abstraction from ethanol could take place from the CH_3 group, the CH_2 group (as shown), or the OH group and, to some extent, hydrogen will be taken from each of the groups. However, both OH and H show a preference for the CH_2 group and most hydrogen abstraction occurs at this point in the molecule. Similar behaviour is found with other organic compounds, with a certain amount of near random hydrogen abstraction from all points in the molecule and, in addition, selective hydrogen abstraction from one, or a small number of, particularly vulnerable carbon atoms.

The overall result of the combination of random and selective hydrogen abstraction is production of a mixture of radiolysis products, with one or two predominating. Mixtures are also produced when pure organic compounds are irradiated, so that irradiation is seldom a useful method for making a particular product in good yield.

Hydrated electrons tend to be more selective in their reactions than OH and H and do not, in fact, react with ethanol or ethene very rapidly. They do react rapidly with hydrogen ions (H^+) in acid solutions, as shown above, and with any oxygen present:

$$e^-_{aq} + O_2 \rightarrow O_2^-$$

Significant in radiation biology, hydrated electrons also react rapidly with thiol (—SH) groups, disulfide groups (—S—S—), and the purine and pyrimidine groups present in DNA and RNA. Hydrogen atoms and hydroxyl radicals also react with these groups, so that they are among the groups most vulnerable to radical attack when organic compounds are irradiated in aqueous solution.

Oxygen molecules have two unpaired electrons and rapidly add to many radicals, including e^-_{aq} and H (but not OH). When aqueous solutions are irradiated in the presence of air, the oxygen present competes with the other dissolved substances for e^-_{aq} and H. The reactions with H and e^-_{aq} give the superoxide ion, O_2^-, and its protonated form, HO_2, respectively; the latter dissociates to O_2^- at pH above 5.

Oxygen will also react with any organic radicals present to give peroxy radicals, for example,

$$\cdot CH_2CH_2OH + O_2 \rightarrow \cdot O_2CH_2CH_2OH$$

interrupting the chain of reactions taking place when oxygen is absent.

The superoxide ion, and organic peroxy radicals, are less reactive than the radicals (e^-_{aq}, H, OH) formed from water directly, and generally react together to form peroxides and other oxidation products:

$$2\,O_2^- + 2H_2O \rightarrow H_2O_2 + O_2 + 2\,OH$$
$$HO_2 + \cdot O_2CH_2CH_2OH \rightarrow H_2O_2 + CH_3CHO + O_2$$

Ethanol irradiated in aqueous solution in the presence of oxygen gives a higher yield of ethanal than in the absence of oxygen.

While generally unreactive, the superoxide ion and peroxy radicals will abstract hydrogen from compounds that have a relatively loosely bound hydrogen atom. Ascorbic acid (vitamin C) and thiols are examples of such compounds, and are among the compounds most susceptible to radical attack when irradiated in aqueous solution in the presence of air. Typical of the reactions that occur are (RSH is a generic thiol)

$$RSH + O_2^- \rightarrow RS\cdot + HO_2^-$$
$$HO_2^- + H^+ \rightarrow H_2O_2$$

Deliberate addition of ascorbic acid or a thiol before irradiation can offer a degree of *protection* to an irradiated aqueous system by scavenging the water radicals before they attack the other molecules present. The protection is sacrificial in the sense that the protecting agent (ascorbic acid or the thiol) is lost in the process. Antioxidants added to foods and other materials to lengthen their storage life are often radical scavengers.

Repair occurs if a thiol, or other hydrogen donor, returns a hydrogen atom to an organic radical formed by loss of H. The following example represents attack on an organic molecule (R′H) by OH and its subsequent repair by a thiol (RSH):

$$R'H + OH \rightarrow R'\cdot + H_2O$$
$$R'\cdot + RSH \rightarrow R'H + RS\cdot$$

From the foregoing it should be clear that the radiation chemistry of even the simplest chemical system is quite complex, and that the radiation chemistry of a biological system will be a great deal more so.

Radiolysis of organic materials

Irradiation of organic materials is an example of *direct* action since the radiation energy is transferred directly to the molecules reacting. Chemical changes produced can generally be traced to breakage of some of the chemical bonds present to give free radicals and the subsequent reactions of the radicals.

Electrons may be excited or lost from any atom in a molecule, but product analysis generally shows that some bonds in the molecule are broken more often than others. This suggests that the adsorbed energy can become localized, at least to some extent, in these bonds.

The relative importance of random bond breaking and breakdown of specific bonds depends on the nature of the compound. When organic iodides are irradiated, for example, the carbon-iodine bond is broken in preference to other bonds in the molecule. However, when straight-chain hydrocarbons such as hexane are irradiated bonds are broken in a more random fashion, as the large number of radiolysis products shows.

Once free radicals have been formed from an organic compound they will react in a similar manner to those generated in aqueous solution. That is, by combination and disproportionation, and by addition of oxygen if it is available. In the absence of oxygen the products generally include unsaturated products from radical disproportionation, and dimeric products formed when two radicals combine to give a larger molecule. The products will cover a range of molecular sizes if irradiation leads to carbon-carbon bonds breaking in a random fashion. Unsaturated compounds may give long-chain polymer molecules as a result of ionic or radical chain reactions.

With proteins and similar polymers the processes of combination and disproportionation lead to crosslinking and to chain breaks, respectively.

When pure liquid ethanol is irradiated the major organic products, CH_3CHO and $HO(CH_2)_4OH$, are the same as those formed in aqueous ethanol solutions because the same intermediate radical, $CH_3\dot{C}HOH$, is produced. However, liquid ethanol gives a larger yield of hydrogen and a larger number of minor products (e.g., CO, CH_4, C_2H_4, C_2H_6, etc.) than a dilute aqueous ethanol solution. Concentrated ethanol solutions, however, give increasing amounts of the minor products as the concentration of ethanol is increased. This is attributed to the direct action of radiation on the ethanol at the higher concentrations; direct action on ethanol in solution gives the same products as irradiation of pure liquid ethanol.

Direct action will be a factor whenever concentrated aqueous solutions are irradiated if the solute contributes a significant fraction of the electrons present in the system.

Aromatic organic compounds suffer less radiation damage than non-aromatic compounds because of their ability to degrade the absorbed energy to heat via low-lying excited states. They can also afford some protection to non-aromatic molecules mixed with them, and non-aromatic portions of a molecule of which they are a part. This type of protection is sometimes referred to as *physical protection* (it is the result of the physical processes of energy transfer and degradation) to distinguish it from the *chemical protection* offered by compounds such as thiols and ascorbic acid in aqueous solution.

Radiation processing (Applied radiation chemistry)

Radiation sources similar to, though more powerful than, those used in radiology and radiation therapy are currently in use in many countries to irradiate polymers, disposable medical supplies, coatings and, to a lesser extent, foodstuffs. Commercial gamma sources can contain up to several MCi (about 100 PBq) cobalt-60 while commercial electron accelerators produce beams of electrons with energies between about 0.15 MeV and 10 MeV at "high" power levels (5 to 300 kW).

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CHAPTER A6

BIOLOGICAL EFFECTS OF RADIATION

A6.1 INTRODUCTION

Not long after the discovery of x-rays in 1895 and of natural radioactivity in 1896, clinical evidence, mainly from effects on the skin indicated that ionizing radiation is harmful to human tissue. Later it was realized that not only is ionizing radiation damaging to most tissue but exposure of the germinal tissue in animals was found to result in effects in the descendants as well.

During almost a century of exploring the uses of ionizing radiation, extensive studies of radiation effects on living species have taken place. These explorations and studies received an enormous impetus following the discovery of nuclear fission in 1939 and the subsequent uses including military to which fission energy was quickly put. It became eviden that the biological effects of ionizing radiation must be studied in order to protect human beings and other species from the harmful effects of radiation while at the same time maximizing the benefits of its use.

Diverse studies in many laboratories throughout the world, while by no means complete, have resulted in a wealth of information concerning the biological effects of radiation, possibly greater than that associated with any other environmental hazard. For radiation protection purposes we are concerned with two types of effects. The first type, non-stochastic effects of radiation, now called deterministic effects, involve the malfunctioning or loss of function of tissues in organs due mainly to cell loss. These effects result from high dose exposures and for them there is a threshold. The second type, stochastic effects, express themselves long after the exposure and include increased risk of cancer and, by implication from studies on animals, hereditary disorders. These stochastic effects appear to have no threshold and may occur after low radiation doses (small fractions of a Gy) even though their frequency is then low. It is currently thought that cancer is the primary effect of concern at low doses.

Deterministic Effects

Deterministic effects in humans can result from general or localized tissue irradiation causing an amount of cell killing that cannot be compensated for by proliferation of viable cells. The resulting loss of cells can cause severe and clinically detectable impairment of function in a tissue or organ. Thus, the severity of the observed effect can be expected to depend upon the dose. There will be a threshold below which the loss of cells is too small to detectably impair tissue or organ function.

In addition to cell killing, radiation can damage tissue in other ways: by interfering with a variety of tissue functions including regulation of cellular components, inflammatory reactions involving modifications and permeability of cells and tissues, mitral migration of cells in developing organs, and indirect functional effects. All of these play a part in the severity of deterministic effects.

Tissues vary in their response to ionizing radiation. Among the most radiosensitive tissues are the ovary and testes, bone marrow and the lens of the eye. In general, the dose frequency relationship for these tissues will be sigmoid in shape when plotted on linear axes, with frequency increasing as the dose increases. Deterministic effects vary with the dose in severity as well as frequency.

As an example of the specific deterministic effect for skin, the threshold for erythema and dry desquamation is about 3-5 gray, the symptoms appearing after about three weeks. Moist desquamation occurs after about 20 gray, blistering after about four weeks. Cell death in the epidermal and dermal layers resulting in tissue necrosis occurs after a dose of about 50 gray and appears after about three weeks.

Whole Body Exposure

Whole body exposures in excess of a few mSv are unusual in medicine, but the effects of higher exposures are described here for completeness.

Acute radiation exposure may be so severe in certain unforeseen circumstances that death may result. Death is generally the result of severe cell depletion in one or more vital organ systems in the body, therefore the dose-response relationship, as observed in cellular studies, is in general relevant. The survival dose relationship is often described by its midpoint, the $LD_{50/60}$, that is the dose at which 50% of the individuals would be expected to die in 60 days. For healthy humans the $LD_{50/60}$ after acute exposure is estimated to be between 3 and 5 gray midline dose (which approximates the marrow dose for low LET penetrating radiation) and the cause of death at this dose is loss of bone marrow function due to loss of bone marrow stem cells.

The Prodromal Radiation Syndrome

The various symptoms making up the human prodromal syndrome vary with respect to the time of onset, maximum severity and duration depending upon size of the dose. With doses of a few thousand cGy all individuals can be expected to show all phases of the syndrome within 5-15 minutes of exposure. Reaction might reach a maximum at 30 minutes. Reaction might reach a maximum by a few days, then gradually diminish in intensity until the prodromal symptoms merged with the universally fatal vascular syndrome or, after a lower dose, with a fatal GI syndrome. The GI symptoms are anorexia, nausea, vomiting, diarrhea, intestinal cramps, salivation, fluid loss, dehydration and weight loss. The neuromuscular symptoms include easy fatigue, apathy or listlessness, sweating, fever, headache and hypotension. All of these signs and symptoms are not seen unless the exposure is in the supralethal range.

Cerebrovascular Syndrome

The total body dose of the order of 100 gray of gamma rays results in death in a matter of hours. At these doses all organ systems will be seriously damaged; the GI and hemopoietic systems will both of course be severely damaged and would fail if the individual lived long enough, but cerebrovascular damage brings death very quickly, so that the consequences of the other system failures is not important.

Gastrointestinal Syndrome

The total body exposure of more than 10 gray of gamma rays leads in most mammals to symptoms characteristic of the GI syndrome, culminating in death some days later (usually between 3 and 10 days). Characteristic symptoms are nausea, vomiting and prolonged diarrhea. Individuals lose their appetites and appear sluggish and lethargic. Prolonged diarrhea extending for several days is regarded as a bad sign, because it indicates the dose received has been more than 10 gray and will prove fatal. After a few days the individual shows signs of dehydration, loss of weight, emaciation and complete exhaustion; death usually occurs in a few days. There is no incidence on record of a human being having survived a dose in excess of 10 gray. The symptoms that appear and the death that follows are attributable principally to the removal of the epithelial lining of the GI tract by radiation.

Prior to Chernobyl there was probably only one example in literature of a human suffering GI death. Several of the fire fighters at Chernobyl, including some who had received bone marrow transplants died of the GI syndrome.

Haematopoietic Syndrome

At a dose of 3-8 gray, death if it occurs is a result of radiation damage to the haematopoietic system. Mitotic active precursor cells are sterilized by the radiation, and the subsequent supply of mature white cells, red cells and platelets is therefore diminished. The time of potential crisis, when the number of circulating cells in the blood reaches a minimum value, is delayed for some weeks. It is only when the mature circulating cells begin to die off and the supply of new cells from the depleted stem cell population is inadequate to replace them, that the full effect of the radiation becomes apparent. As a consequence of the Chernobyl accident, 203 operating personnel, firemen and emergency workers were hospitalized suffering from the early radiation syndrome, having received doses in excess of 1 gray. About 30 of these who had received doses in excess of 4 gray died. The remainder recovered with conservative medical treatments.

Stochastic Effects

"Stochastic" effects are those for which the probability of an effect occurring, rather than its severity, is regarded as a function of dose, without threshold.

At the dose range involved in radiation protection, hereditary effects are regarded as being stochastic. Some somatic effects are stochastic; of these, carcinogenesis is considered to be the chief somatic risk of irradiation at low doses and therefore the main problem in radiation protection.

Radiation Carcinogenesis

The mechanisms by which radiation carcinogenesis occurs are by no means clear. An attractive theory, which is often suggested, is that carcinogenesis results from somatic mutation in a normal tissue cell. It is thought that the induction of cancer involves at least two stages. In initiation a low dose of carcinogen acts on DNA perhaps to form an oncogene. Subsequently this oncogene may be activated by a promoter, which may be a chemical which is not carcinogenic by itself but acts through free radical processes.

At low doses and dose rates, which is the area of concern for radiation protection in medicine, on average less than one event per sensitive cell occurs. As discussed earlier it is assumed that there is no threshold for the induction of the molecular changes in DNA that result in malignant transformation.

New values of the risk estimates from ionizing radiation were recently published (ICRP 60). The main focus of the document is on the risk of fatal cancer from radiation. The new (1986) risk values take into account the new dosimetry of the Japanese bombs, a longer follow-up of the survivors, and, most of all, a change in the predictive model used to fit the epidemiological data. The new data fit more closely a relative risk model rather than an additive model used in ICRP 26. In a relative risk model cancer induction is related to the natural incidence, so for the atomic bomb survivors many cancers have still to be expressed as the population ages.

The fatal cancer risk estimate for the whole population is now 5% Sv^{-1} up from 1.25 in ICRP 26.

A6.2 The Genetic Effects Of Radiation

There is substantial evidence that radiation-induced genetic changes in lower organisms are truly stochastic, but no reliable data exists in the case of humans. In the case of the atomic bomb survivors 70,000 pregnancies were studied, although less than a third of these received more than 0.01 Gy. There was a trend to a possible increase in potential genetic effects at higher doses, with an indication that the dose to double the natural mutation rate was in the region of 1 Sv (95% confidence limits 0.2 Sv and infinity). Because the human data are not statistically reliable other methods have been used to estimate the damage from radiation.

For simplicity let us consider three general classes of diseases which have a strong genetic component:

monogenic, which includes dominant, recessive and cross-linked mutations of a single gene

chromosomal, which are due to changes in the structure of a number of chromosomes in the germ cell, and

multigenic(multifactorial), where the disease is associated with the interaction of a considerable number of genes and of the environment. This group includes many multigenic disorders such as adult onset diabetes, glaucoma, essential hypertension, MI, varicose veins, asthma, GI ulcers and rheumatoid arthritis, which normally occur in adulthood and for which there is good evidence for a heritable component.

Normal Incidence of Genetic Disorders in Humans
Incidence per 100 Live Births

Category	UNSCEAR 88	BEIR 90
Monogenic	1.3	1.3
Chromosomal	0.4	0.4
Multigenic	66	120

(persons can suffer from more than one disease)

From information on radiation effects on rodents and primates estimates have been made of the radiation-induced probability of genetic diseases and the doubling dose has been taken as 1 Sv. The latest of these gives a value 0.1×10^{-2} for the first generation and 1.0×10^{-2} per Sv for all generations (ICRP 1990).

Radiation Effects on the Embryo and Foetus

It has long been recognized that the embryo and foetus might constitute a specially radiosensitive group because of the great amount of actively dividing cell populations. Much of the data comes from experimentally irradiated rodent populations, but special human populations such as the atomic bomb survivors and those irradiated for medical purposes while pregnant have been intensively studied.

Pre-implantation commences with fertilization and after several divisions, and lasts to the beginning of implantation (4-5 days). The embryonic period commences with implantation, which itself takes about 14 days, until complete placental blood flow is established. Germ layer formation occurs between the end of the first and third weeks. The fourth to eighth weeks are especially important as all major organ system start to appear, and are 90% present at the end of this period. The foetal period continues this progression as the organs grow, differentiate, develop and mature to birth.

Congenital Abnormalities

The process of development from conceptus to birth is highly complex and in many cases unsuccessful. It is thought that the loss of human embryos is approximately 40% between conception and 20 weeks, with 10% occurring between 10 and 18 weeks. Most occur before pregnancy has been diagnosed or a period missed. About 15% of recognized conceptions will lead to spontaneous abortions during early and intermediate foetal development, and one third of these foetuses carry some kind of chromosomal abnormality. About 6% of live born children have abnormalities.

Prenatal and Neonatal Death

Evidence for prenatal or neonatal death caused by radiation in humans is sparse, and estimates of the relevant LD_{50} have been extrapolated from rodent data.

Values that have been suggested for LD_{50} are:

Time After Conception	LD_{50}
1 week	1.0 Sv
5-7 weeks	1.5 Sv
21 weeks on	3.0 Sv

Severe Mental Retardation

Data from the atomic bomb survivors in a study of 1600 children exposed in utero has shown about 30 cases of severe mental retardation (normal rate 0.8%). The most sensitive gestational period was 8-15 weeks, and in this period the fraction of those retarded increased by 0.4 per Sv. For the period 16-25 weeks the rate was 0.1 per Sv. There appeared to be a threshold of around 0.2 Sv.

Reduced Intelligence

Mental impairment of lower severity is also apparent in children exposed in utero. This is a dose related decrease in IQ of about 30 units per Sv again in the 8-15 week period.

Cancer Induction

Irradiated foetuses seem to be more sensitive to the induction of cancer before the age of 10. There is disagreement between the Japanese data and that derived from prenatal irradiation for medical purposes. Current best estimates are 2.5×10^{-2} per Sv for leukemia and 3.5×10^{-2} per Sv for other cancers.

Patient Counselling

At some point you will be faced with a pregnant patient who has been inadvertently exposed. What do you do to reassure this patient?

In almost every case nowadays the radiation dose to the foetus poses only a trivial risk in real terms both from a stochastic and non-stochastic viewpoint. It is important to compare any possible risk with the "normal" risks of childbirth. Only in exceptional cases will it be necessary for detailed calculations to be made.

Dose* To Conceptus mSv (mrem)	Child With No Malformation (%)	Child Will Not Develop Cancer (Percent)	Child Will Not Develop Cancer Or Have A Malformation (Percent)
0. (0)	96	99.93	95.93
0.5 (50)	95.999	99.927	95.928
1.0 (100)	95.998	99.921	95.922
2.5 (250)	95.995	99.908	95.91
5.0 (500)	95.99	99.89	95.88
10.0(1000)	95.98	99.84	95.83

Based on Wagner LK, Haymal LA: Pregnancy in women radiologists. Radiol 145:559-562, 1982.
**Refers to absorbed dose above natural background. This table assumes conservative risk estimates, and it is possible that there is no added risk.*

How We View Risk

"Risk" is a very difficult subject to talk about. In the scientific sense risk combines two ideas - the probability that an event will occur, and the severity of the effect of that occurrence.

So let us think for the moment just about the probability or the likelihood that something will happen. For those of us who are pessimistic, we may see the whole process of living as a constant battle with the possibility of death or dismemberment. We may fall out of bed in the morning, we may cut our throats while shaving, or fall over on the driveway. We may get killed in an accident on the way to work and we may prick ourselves with a needle carrying an infectious disease while we are working in a laboratory.

However most of us come to accept that there two sides to the question of risk. In most cases we accept that the benefits from our actions far outweigh any probability of risk to our persons.

Now in some cases we accept personal responsibility for accepting certain risks, such as whether we choose to smoke cigarettes or whether we chose to climb mountains or go hang-gliding.

Often, however, it is difficult for us to appreciate what a risk is - for example, we have a chance of one in ten million of individually being killed by lightning. As lightning generally kills only one person at a time, the risk to each of us is extremely low, and we treat it as virtually negligible. Because we live in a society and have to function together, there are certain risks that have been regulated by society as a whole, with the aim of securing benefits for all. Thus there are laws and regulations to reduce traffic accidents. The following table gives the general levels of fatal risk per annum for some risks of living.

Levels of Fatal Risk

1 in 1000	Risk of death in high risk groups in industries such as mining
1 in 10,000	General risk of death in traffic accidents
1 in 100,000	Risk of death in an accident at work in the safest industries
1 in 1 million	General risk of death in a fire or explosion at home
1 in 10 million	Risk of death by lightning

The Acceptability of Risk

One of the problems of talking about risk is that there can never be a zero risk situation for anything that we do. The words safe and safety are therefore only relative. However, we may be able to define the levels of safety or risk which most people would find acceptable or safe enough.

In general, it is felt that a risk of one in a million is a very low risk and is to all intents and purposes negligible. On the other hand, a risk of one in a thousand is probably verging on the unacceptable for an occupational risk.

The average exposure amongst radiation workers in Canada is about 1 mSv/year. From the ICRP risk figures we can see that the extra imposed risk of fatal cancer at an annual exposure of 1 mSv/year would be about five in a hundred thousand.

However, we all may feel cautious about adding involuntary risk to the ones which we cannot help or which we decide for ourselves. Further, people demand that such extra risks be reduced to very low levels indeed compared to the other risks that they accept for themselves. In dealing with hospital personnel regarding radiation exposure we often have to address the "outrage" over this apparently unwarranted risk rather than the numerical value of the risk.

In so far as patients are concerned, it is the duty of the physician to weigh the risks against the medical benefits of any diagnostic or therapeutic radiation exposure.

CHAPTER A7
ADVISORY BODIES

A7.1 INTRODUCTION

Canadian radiation regulations are formulated after deliberate evaluation of recommendations made by the International Commission on Radiological Protection, and similar respected scientific bodies. Most countries follow the same route to their own national regulations.

This chapter contains information on some of these scientific organizations, and their publications.

A7.2 THE I.C.R.P.

In the early 1920's, the British X-ray and Radium Committee and the American Roentgen Ray Society proposed general radiation protection recommendations on the basis of avoiding acute effects (e.g. erythema); and in 1925, at the First International Congress of Radiology, the need for quantifying exposure was recognized. As a result, in 1928, the International Commission on x-ray and Radium Protection was established. (Rolf Sievert of Sweden was a founding member). In 1950, in line with the more general applications of ionizing radiations and radioactive materials, the commission was renamed as the **International Commission on Radiological Protection** (ICRP).

The Actual Commission consists of a chairman, and not more than twelve other members. selected from nominations submitted to the International Congress of Radiology. Members are chosen on the basis of their recognized work in medical radiology, radiation protection, physics, health physics, biology, genetics, biochemistry and biophysics, with regard to an appropriate balance of expertise rather than to nationality.

Other individuals may be invited to Commission meetings to give special technical advice, and special committees may be established to deal with particular topics. The committees in turn usually form task groups, for which they call upon the services of numerous experts who are not members of the Commission. Decisions of the ICRP are made by majority vote of the members, with provision for minority opinions.

The Commission has an official relationship with the World Health Organization (WHO) and the International Atomic Energy Agency (IAEA), and maintains close working relationships with the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), and several other international bodies.

The ICRP publishes its basic philosophy and recommendations about once every ten years. The last such report was ICRP Publication 60 in 1990, (reviewed below). The publication of one of these "Recommendations of the ICRP" reports is an important event in the radiation protection world, affecting such fundamental parameters as the annual maximum permitted dose (MPD), and often changing the way radiation protection is practiced for the next decade or so.

In addition to these basic reports, the ICRP also convenes "expert committees" to investigate and report on a variety of specific areas, such as Protection of the Patient in Nuclear Medicine (ICRP 52), or Radiation Dose to Patients from Radiopharmaceuticals (ICRP 53).

All Reports and Recommendations of the ICRP are available as Annals of the ICRP. Individual reports may be purchased, or a subscription taken out for the whole series. Publication is by **Pergamon Press, Maxwell House, Fairview Park, Elmsford, N.Y. 10523. Tel: (416) 283-2914** (this Toronto phone number rings through to New York), **or (914) 592-7700**.

Brief reviews of some of the more medically oriented ICRP reports follow.

ICRP Publication 28: The Principles and General Procedures for Handling Emergency and Accidental Exposures of Workers. December 1978; 22 pp.

Every radiation safety program should contain a plan for action in the event of abnormal exposures i.e. exposures in excess of the limits recommended for normal practice. This report discusses such actions. It does not deal with the subsequent medical treatment for the effects of radiation exposure, nor with actions required to bring the source of exposure under control. It does deal with the assessment of the severity of the exposure, the immediate medical care of injuries not caused by radiation, but possibly complicated by its presence (e.g. treatment of contaminated wounds), administrative actions following the abnormal exposure, the organization of medical services in anticipation of abnormal exposures, and the screening of workers to determine which have been exposed. Appendices contain tables of symptoms of "Radiation sickness", and action check lists.

Without saying so, this report appears directed to a Chernobyl type disaster, but any RSO formulating a plan for the emergency admission of patients involved in radiation accidents would find it useful.

ICRP Publication 33: Protection Against Ionizing Radiation from External Sources used in Medicine. August 1982; 74 pp.

With the introduction of the new concepts contained in ICRP publication 26, many previous reports had to be updated. This report supersedes ICRP publications 15 and 21, and deals with radiology and radiotherapy. It therefore contains mainly information of ancillary interest, but includes summaries of parts of ICRP 26 which are also applicable to the use of radionuclides.

ICRP Publication 35: General Principles of Monitoring for Radiation Protection of Workers. January 1983; 40 pp.

This report deals with the why's and wherefore's of all types of monitoring: area monitoring (measuring radiation levels), monitoring for surface and skin contamination, air monitoring, personal monitoring, and internal contamination monitoring. It discusses the interpretation of results, and limits and reference levels. As with all ICRP reports, details are deliberately omitted, and attention focussed on concepts, definitions and philosophies.

ICRP Publication 38: Radionuclide Transformations: Energy and Intensity of Emissions. December 1983; 1200 pp.

A large volume of tables and decay scheme diagrams detailing radionuclide emissions. A valuable acquisition for your local university library.

ICRP Publication 51: Data for use in Protection against External Radiation. January 1988; 148 pp.

This report is intended to supply information and advice on the relationships between radiometric, dosimetric and radiation protection quantities for external radiation and on their practical utilization. A radiometric quantity is one that defines the radiation, particle flux for example. A dosimetric quantity, such as absorbed dose, arises from the interaction of the radiation with matter. A radiation

safety quantity, such as effective dose equivalent, is used as an approximate indication of detriment, and to express basic limits for protecting persons.

This report, therefore, discusses the difficult problem of converting physical radiation measurements into effective dose equivalent, for different exposed organs, different types of radiation, and a variety of irradiation geometries. While it may be of limited immediate use in routine hospital radiation safety, it will afford the dedicated RSO a deeper insight into the meaning and relationships of the various radiation quantities.

ICRP Publication 52: Protection of the Patient in Nuclear Medicine. February 1988; 46 pp.

This publication relates to exposures to patients from the administration of radiopharmaceuticals for diagnostic, therapeutic and research purposes, including recommendations on the protection of the patient's family. It begins with a discussion of the responsibilities of the various members of the medical team: referring physician, nuclear medicine physician, medical physicist, technologist etc. It advises on factors that influence absorbed dose from different types of nuclear medicine examinations and indicates ways by which radiation risks can be minimized without detriment to intended medical benefits. It should be read in conjunction with ICRP 53 (below).

ICRP Publication 53: Radiation Dose to Patients from Radiopharmaceuticals. March 1988; 386 pp.

The first 30 pages of this report contain a review of the absorbed fraction calculation method, comments on biokinetic models, a discussion of the role of effective dose equivalent in nuclear medicine, and advice on the problem of impurities in radiopharmaceuticals.

The rest of the report consists of tables of absorbed dose and effective dose equivalent for about 120 radiopharmaceuticals.

ICRP Publication 54: Individual Monitoring for Intakes of Radionuclides by Workers: Design and Interpretation. August 1988; 200 pp.

This publication discusses the design of individual monitoring programs and the interpretation of the results of measurements of intakes of radionuclides. The most common current example of such a program is the mandatory bioassay program for radioiodine intake.

ICRP Publication 57: (Annals of the ICRP Vol. 20 No. 3) Radiological Protection of the Worker in Medicine and Dentistry. 1990.

The final updating of ICRP Publication 25 (which is now withdrawn from circulation), containing more-than-usual detail (for ICRP) on radiation protection in medical establishments.

ICRP Publication 60: (Annals of the ICRP Vol. 21 No. 1-3) Recommendations of the International Commission on Radiological Protection. 1991; 201 pp.

The Commission has revised its 1977 comprehensive radiation protection recommendations embodied in its Publication 26. This ICRP Publication 60 revision of the Commission's recommendations is stated to have three aims:

a) to take account of new biological information and of trends in the setting of safety standards,

b) to improved the presentation of the recommendations and

c) to maintain as much stability in the recommendations as is consistent with the new information.

The Commission's recommendation are given in the main text supported by three annexes that provide further details and explanations.

The Users' Edition, 1992 is a slimmer booklet (83 pp.) published at a more reasonable price. It contains the full text of recommendations without its rationale.

ICRP Publication 61: (Annals of the ICRP Vol. 21 No. 4) Annual Limits on Intake of Radionuclides by Workers Based on the 1990 Recommendations. 1991; 41 pp.

This document contains a set of calculated secondary limits for the intake of radionuclides by occupationally exposed workers. These annual limits on intake (ALI), for both the oral and inhalation pathways, are based on the use of the radiation and tissue-weighting factors recommended in the International Commission on Radiological Protection's (ICRP) Publication 60 (1991) and assume an average annual limit on the committed effective dose of 0.02 Sv. The document contains calculated results for all radionuclides considered in ICRP Publication 30 (1979) and includes changes found in other ICRP documents, e.g., ICRP Publication 48 (1986). No values are given for the derived air concentration (DAC).

ICRP Publication 62 (Annals of the ICRP Vol. 22 No. 3) Radiological Protection in Biomedical Research.1993; 18 pp.

The Commission published this report as a review of risks and benefits of research involving human subjects. Changes in risk estimates (ICRP 60) and more extensive information about the effect of age at time of exposure, gender differences, and consequences of in utero exposure prompted the review. The objectives of the report are to provide advice to :

a) individuals planning research involving radiation exposure of human subjects,

b) regulatory and other authoritative bodies involved in issuing rules of conduct,

c) individuals or groups involved in the assessment of specific research projects, and

d) patients and normal volunteers to assist them in understanding radiological risks and in making appropriate decisions related to their participation in clinical investigations.

ICRP Publication 73: Radiological Protection and Safety in Medicine. 1996.

The report clarifies how the recommended system of radiological protection as described in the 1990 Recommendations of ICRP should be applied in medicine. The report is addressed principally to physicians and physicists directly engaged in medical radiology, including diagnosis in medicine and dentistry, nuclear medicine, and radiotherapy; to those responsible for the management of institutions operating in these fields; and to international regulatory and advisory bodies.

ICRP Publication 75: General Principles for the Radiation Protection of Workers. 1997

ICRP *Publication 75* reports comprehensively on the principles for the protection of workers from ionising radiation. It develops guidance on the implementation of the principles in the 1990 Recommendations of ICRP *(Publication 60)*, including the concepts of constraint and reference levels. The report discusses the management of occupational exposure in normal and emergency situations, in industrial and medical contexts, and with respect to natural sources of radiation, including radon, at work. Health surveillance of workers and the management of overexposed individuals are considered. The report updates ICRP *Publication 28* with respect to principles and procedures for handling emergency and accidental exposures of workers, and, by laying out the principles of monitoring for external radiation, completely replaces ICRP *Publication 35*. Monitoring for radionuclide contamination is also discussed.

ICRP Publication 77: Radiological Protection Policy for the Disposal of Radioactive Waste.

The report describes the general policy of ICRP concerning radioactive waste disposal, including releases into the environment. It reaffirms the Commission's policy of radiological protection, in particular its policy on public exposure, and aims to

clarify the practical application of that policy to the disposal of radioactive waste. The report discusses the justification of a practice, the optimisation of protection, the use of collective dose assessed over long distances and times, the implications of potential exposure, and the distinction between practices and intervention.

A7.3 THE A.C.R.P., A.C.N.S. AND G.M.A.

Since the 1950's, the Atomic Energy Control Board (AECB) has made use of advisory committees of independent experts to assist it in its decision-making process.

In 1979 the Board restructured the organization of these consultative groups, resulting in the creation of two senior level scientific committees charged with providing the Board with independent advice on principles, standards and general practices related to radiation protection and the safety of nuclear facilities. The two committees are the Advisory Committee on Radiological Protection (ACRP), formed in 1979, and the Advisory Committee on Nuclear Safety (ACNS), which was established a year later.

A third body of advisers known as the Group of Medical Advisers (GMA), is composed of medical practitioners from provinces in which regulated nuclear activities are carried out. The GMA made recommendations to the Board respecting, inter alia, the medical examination of atomic radiation workers, medical surveillance required as a result of overexposures, and medical aspects of emergency plans. In 1999 the GMA was combined with the Advisory Committee on Radiological Protection, which was augmented with additional medical members.

From time to time the committees and the GMA issues reports which are normally published by the AECB and catalogued within the AECB's public document system. These reports, bound with a distinctive cover, carry both a group-designated reference number, e.g. ACRP-1 or ACNS-1 or GMA-1, and an AECB reference number in the "INFO" series. The reports generally fall into two broad categories:

(i) recommendations to the AECB on a particular technical or medical topic, and
(ii) background studies.

Unless specifically stated otherwise, publication by the AECB of a report prepared by a Committee or by the Group of Medical Advisers does not imply endorsement by the Board of the content, nor acceptance of any recommendations made therein.

Reports published by the Advisory Committee on Radiological Protection

ACRP-1: Risk Estimates for Exposure to Alpha Emitters, July 1982 (AECB INFO-0090).

ACRP-2: Risks of Low-LET Radiation as Given in BEIR-III and Previous Reports, July 1982 (AECB INFO-0091).

ACRP-3: Recommendation on Criteria for the Protection of the Public in the Event of a Nuclear Emergency, July 1983 (AECB INFO-0107).

ACRP-4: Interim Recommendations on the Revisions to the AEC Regulations Pertaining to Limits for the Exposure of Persons to Ionizing Radiation, September 1993 - submitted to the AECB for its use only (not published).

ACRP-5: Assessment for Medico-Legal Purposes of the Contribution of Occupational or Other Defined Exposure to Ionizing Radiation as Causative Agent in Individuals Suffering From or Having Died of Cancer, February 1984 (AECB INFO-0120).

ACRP-6: Harm to Offspring of Women of Childbearing Age Employed in the Nuclear Industry, February 1984 (AECB INFO-0121).

ACRP-7: Report on the Derived Working Limits for Surface Contamination, July 1993 (AECB INFO-0451).

ACRP-8: Recommendations on Exposure Limits for Uranium Miners - not published.

ACRP-9: Radiation Doses from Medical Diagnostic Procedures in Canada, March 1997 (AECB INFO-0670).

ACRP-10: Toxicity and Dosimetry of Tritium - A Review, January 1991 (AECB INFO-0377 (E)).

ACRP-11: Basic Principles of Radiation Protection in Canada, March 1990 (AECB INFO-0340(E)).

ACRP-12: Radiological Hazards to Uranium Miners, May 1990 (AECB INFO-0352).

ACRP-13: A Review of Radiation Risk Estimates, June 1991 (AECB INFO-0389).

ACRP-14: Principles for the Regulation of Carbon-14 in Effluents from Canadian Nuclear Facilities - In preparation.

ACRP-15: The Management of Workers Occupationally Exposed to Ionizing Radiation, October 1993 (AECB INFO-0484(E)).

ACRP-16: Elements of a Radiation Protection Program for Medical and Academic Institutions, September 1997 (AECB INFO-0685).

ACRP-17: Report of External Dosimetry in Canada, December 1995 (AECB INFO-0621).

ACRP-18: Biological Effects of Low Doses of Radiation at Low Dose Rates, May 1996 (AECB INFO-0654).

ACRP-19: Radiation Monitoring of Uranium Workers, December 1998 (AECB INFO-0699).

Reports published by the Advisory Committee on Nuclear Safety

ACNS-1: Recommendations on AECB Draft Licensing Guides Nos. 40, 41, 42, June 1981 (AECB INFO-0054).

ACNS-2: A Proposed Statement on Safety Objectives for Nuclear Activities in Canada, April 1982 (AECB INFO-0055/Rev.1)

ACNS-3: A Report on the Use of Programmable Digital Computers in the Shutdown Systems of the Darlington G.S., June 1981 (AECB INFO-0056).

ACNS-4: Recommended General Safety Requirements for Nuclear Power Plants, June 1983 (AECB INFO-0116).

ACNS - 5 & 5A: Emergency Core Cooling Systems in Candu Nuclear Power Plants, December 1981 / June 1982 (AECB INFO-0068 / Rev. 1).

ACNS-6: A Report on Human Factors in Nuclear Safety, March 1983 (AECB INFO-0225).

ACNS-7: Report on the AECB Consultative Document C-70: "The Use of Fault Trees in Licensing Submissions", January 1984 (AECB INFO-0126).

ACNS-8: A Report on the Public Perception of Risk, July 1986 (AECB INFO-0240(E)).

ACNS-9: A Proposed Regulatory Policy Statement on Human Factors Requirements in the Design and Operation of Canadian Nuclear Facilities, October 1986 (AECB INFO-0226).

ACNS-10: Alternative Electrical Energy Systems - A Comparison of the Risks of Occupational and Public Fatalities, March 1989 (AECB INFO-0299(E)).

ACNS-11: Principles and Guidelines for Radioactive Waste Disposal Facilities, June 1988 (AECB INFO-0294).

ACNS-12: General Principles Underlying the Decommissioning of Nuclear Facilities, March 1988 (AECB INFO-0273).

ACNS-13: Recommended Principles for the Commissioning of Nuclear Power Plants, September 1989 (AECB INFO-0317).

ACNS-14: Recommended Safety Objectives, Principles and Requirements for Mini-Reactors, May 1991 (AECB INFO-0388(E)).

ACNS-15: Recommendations and Comments on the Report of the Ontario Nuclear Safety Review, December 1988 (AECB INFO-0339).

ACNS-16: Assessment of IAEA Safety Series Report No. 75-INSAG-3 "Basic Safety Principles for Nuclear Power Plants" January 1989 (AECB INFO-0316).

ACNS-17: Improving the Safety Culture of Human Organizations: A Proposal for Institutional Quality Assurance, February 1990 (AECB INFO-0359).

ACNS-18: Review of Canadian Programs for Radioactive Waste Management and Disposal, December 1992 (AECB INFO-0449).

ACNS-19: Recommendations Relating to Safety-Critical Real-Time Software in Nuclear Power Plants, January 1992 (AECB INFO-0404).

ACNS-20: Proposed Quantitative Approach to Safety for Nuclear Power Plants in Canada, July 1995 (AECB INFO-0568).

ACNS-21: Recommended General Safety Principles and Requirements for Nuclear Power Plants in Canada - In preparation.

ACNS-22: Feedback of Operating Experience in Nuclear Power Plants, June 1995 (AECB INFO-0567.

ACNS-23: Transport of Radioactive Material in Canada, March 1998 (AECB INFO-0683).

ACNS-24: Basic Principles for Regulating Nuclear Activities, September 1996 (AECB INFO-0660).

ACNS-25: Nuclear Power Plant Life Management in Canada - In preparation.

Reports published jointly by the Advisory Committee on Radiological Protection and the Advisory Committee on Nuclear Safety

AC-1: Recommended *De Minimis* Radiation Dose Rates for Canada, July 1990 (AECB INFO-0355).

AC-2: Application of the ALARA Process in the Regulation of Nuclear Activities, May 1991 (AECB INFO-0387(E)).

AC-3: Nuclear Emergency Preparedness in Canada, March 1993 (AECB INFO-0443).

AC-4: A Survey of Research Programs in Radiation Protection in Canada, July 1995 (AECB INFO-0547).

AC-5: Assessment and Management of Cancer Risks from Radiological and Chemical Hazards, April 1998 (AECB INFO-0684). This report was prepared by a joint working group with representatives from the AECB Advisory Committees, the AECB, Health Canada, and the Ontario Ministry of Environment and Energy.

Reports published by the Group of Medical Advisers

GMA-1: Guidelines for the Medical Surveillance of Atomic Radiation Workers, November 1991 (AECB INFO-0402). (Note: This report is superseded by GMA-8).

GMA-2: Guidelines on Radiation Exposure Levels for Emergency Situations, July 1992 (AECB INFO-0442).

GMA-3: Guidelines on Hospital Emergency Plans for the Management of Minor Radiation Accidents, December 1992 (AECB INFO-0427(E)).

GMA-4: Guidelines on the Management of Patients Treated With I-131, March 1993 (AECB INFO-0442). Reprinted in May 1993.

GMA-5: Guidelines for Research on Human Subjects Using Ionizing Radiation, December 1993 (AECB INFO-0491).

GMA-6: Guidelines on Counselling After Occupational Overexposure to Ionizing Radiation, May 1993 (AECB INFO-0450).

GMA-7: Guidelines on the Medical Management of Tritiated Water Overexposures, September 1995 (AECB INFO-0586).

GMA-8: Revised Guidelines for the Medical Surveillance of Radiation Workers, May 1993 (AECB INFO-0452). This supersedes the GMA-1 report.

GMA-9: Guidelines on the Use of Stable Iodine As a Prophylactic Measure During Nuclear Emergencies, September 1995 (AECB INFO-0587).

GMA-10: The Observed and Predicted Health Effects of the Chernobyl Accident - March 1996 (AECB INFO-0623).

GMA-11: Principles of the Management of Radionuclide Therapies - In preparation. (This report is being prepared by a joint working group with the ACRP and will be published as report AC-9).

GMA-12: Guide to Ionizing Radiation Exposures for the Occupational Physician. A draft version of this Internet document was posted on the AECB web site in November 1998.

GMA-13: Review of Quality Assurance in Radiation Therapy, May 1998 (AECB INFO-0687).

A7.4 THE I.C.R.U.

The **International Commission on Radiation Units and Measurements**, founded in 1925, makes recommendations on quantities and units relating to radiation and radioactivity, and on procedures for the measurement of these quantities in clinical radiology and radiobiology, and in the field of radiation protection, where they work closely with the ICRP. Their reports on radiation quantities and units contain the "gold standard" definitions, and, of necessity, are very precise and mathematical, but worth reading.

Further information may be obtained from:

ICRU publications, 7910 Woodmont Ave., Suite 800, Bethesda, Md. 20814, U.S.A.

ICRU Report 39: Determination of Dose Equivalents Resulting from External Radiation Sources (1985).

ICRU Report 43: Determination of Dose Equivalents from External Radiation Sources - Part 2 (1988).

ICRU Report 47: Measurement of Dose Equivalents from External Photon and Electron Radiations (1992).

ICRU Report 51: Quantities and Units in Radiation Protection Dosimetry (1993).

A7.5 THE I.A.E.A.

The International Atomic Energy Agency was formed in July, 1957, and is an independent intergovernmental organization within the United Nations System. The Agency currently has 113 member states (Canada was a founding member), and its objective is: "To accelerate and enlarge the contribution of atomic energy to peace, health, and prosperity throughout the world and to ensure so far as it is able that assistance provided by it, or at its request or under its supervision or control, is not used in such a way as to further any military purpose."

The IAEA Bulletin can be obtained, free of charge (so far), by writing to the Editor at:

The International Atomic Energy Agency, Wagramerstrasse 5, P.O. Box 100, A-1400 Vienna. It often contains interesting articles on the practice of nuclear medicine and other uses of radionuclides, especially in "third world" countries, and advertisements for jobs.

A7.6 THE N.C.R.P.

In 1928, when the ICRP was being founded, there was some rivalry between the American Roentgen Ray Society and the Radiological Society of North America as to whose differing views on radiation protection should be presented at the international level as representative of the U.S.A. Largely due to the influence of Lauriston S. Taylor of the National Bureau of Standards (NBS), one of the "grand old men" of radiation protection, the two societies agreed to consolidate their protection activities into a single committee managed by the NBS, but not subject to its control. The American Medical Association, the American Radium Society and the x-ray equipment manufacturers also appointed representatives to the new committee, called the "Advisory Committee on x-ray and Radium Protection", and chaired by L.S. Taylor.

Almost immediately they began publishing handbooks. NBS Handbook 15 was published in 1931, and NBS Handbook 20 in 1936. The latter contained the first dose limit, or "tolerance dose" as it was then called (0.1 r per week), which was replace by the "maximum permissible dose" in 1947.

In 1947, the name was changed to the "National Committee on Radiation Protection", and in 1957 the words "and Measurements" were appended. In 1964, when they were granted a U.S. Congressional Charter, the name became the **"National Council on Radiation Protection and Measurements"**, still abbreviated to NCRP.

Radiation regulations in the U.S. are only slightly different from those in Canada, and NCRP reports therefore are very applicable to the Canadian situation, and their recommendations might be considered "good practice", even if they sometimes go one step beyond what is required by Canadian regulations. NCRP reports also tend to be more practically oriented, and more readable than ICRP reports. For these reasons, some of the relevant NCRP reports are reviewed below. Those marked with an asterisk * have some direct relevance to radionuclide radiation safety, the others are more in the area of "professional curiosity" only. A complete list of reports with prices may be obtained from:

NCRP Publications, 7910 Woodmont Ave, Suite 800, Bethesda, Md. 20814

NCRP Report No. 37: Precautions in the management of Patients who have Received Therapeutic Amounts of Radionuclides. First issued October 1970, latest reprint March 1978: 62 pp.

A very useful and relevant document, containing facts and figures and recommended procedures for handling radionuclide therapies, including I-131 therapies, release of patient from hospital, emergency surgery or death of radioactive patients, and burial or cremation.

NCRP Report No. 54 Medical Radiation Exposure of Pregnant and Potentially Pregnant Women. Issued July 1977, reprinted May 1985; 35 pp.

A useful discussion of, and recommendations on, the subject of diagnostic examinations of pregnant or potentially pregnant women, including a decision guide for scheduling or postponing such examinations and the estimated dose to the ovaries from various x-ray procedures.

NCRP Report No. 65: Management of Persons Accidentally Contaminated with Radionuclides. First issued April 1980, latest reprint May 1987; 206 pp.

This report is intended for the physician faced with the management of accidents involving radionuclides. The preface stresses that it is a guide only, for the initial stages of management, and cannot be used as a substitute for the knowledge and judgment of the responsible physician, or for the information and advice available from specialists who have had actual experience with such cases.

However, much of the initial part of the report concerns tasks which the hospital RSO may be required to perform, such as contamination measurements, identification of the contaminating radionuclide, bioassay sampling etc. Appendix A describes the U.S. Interagency Radiological Assistance Plan (RAP), which does not include Canada.

NCRP Report No. 70: Nuclear Medicine - Factors Influencing the Choice and use of Radionuclides in Diagnosis and Therapy. Issued June 1982; 171 pp.

What makes a good nuclear medicine radionuclide? Half life, energy and type of emissions, dose to the patient, availability, price, and a whole lot more. Read all about it in this report. It includes dose information on commonly used radiopharmaceuticals and guidelines for the clinical evaluation of radiopharmaceutical drugs by a subcommittee of the U.S. Federal Drug Administration.

NCRP Report No. 73: Protection in Nuclear Medicine and Ultrasound Diagnostic Procedures in Children. Issued December 1983; 81 pp.

How to reduce the radiation dose to child patients, and still achieve the maximum useful diagnostic information. The report contains tables of pediatric radiation doses, and a brief discussion of the biological effects of ultrasound. (NCRP Report No. 74 has a more detailed discussion of ultrasound)

NCRP Report No. 80: Induction of Thyroid Cancer by Ionizing Radiation. Issued March 1985; 93 pp.

A review of the scientific evidence for the induction of thyroid cancer due to external and internal (e.g. radioiodine) radiation, ending with a formula for calculating the risk from a known dose.

NCRP Report No. 82: S.I. Units in Radiation Protection and Measurements. Issued August 1985; 64 pp.

A brief but interesting history of the evolution of radiation units, followed by a more detailed discussion of the S.I. units used in radiation protection. It also contains discussion of the adoption of S.I. units by the U.S., which is not relevant to the Canadian situation. Appendices contain conversion tables. A useful explanation of radiation protection units.

NCRP Report No. 83: The Experimental Basis for Absorbed-Dose Calculations in Medical Uses of Radionuclides. Issued September 1985; 109 pp.

A review of the methods used to estimate the radiation dose from internally deposited radionuclides, especially in nuclear medicine. The emphasis of this report is on the comparison of the results of direct measurements to the results of

computer calculations (such as the MIRD method). A series of recommendations is made for the improvement of the accuracy of internal dosimetry results, including the suggestion that dose measurements be made on nuclear medicine patients for comparison with calculated values. A useful insight into internal dosimetry and its current accuracy.

NCRP Report No. 84: General Concepts for the Dosimetry of Internally Deposited Radionuclides. Issued September 1985; 109 pp.

A broad review of the philosophy and methodology of limiting radiation dose from internal emitters. Primary concepts introduced by ICRP 26 (e.g. committed dose equivalent) are reviewed and explained, and followed up by an NCRP evaluation and recommendation on the practical use of the concept for radiation protection.

NCRP Report No. 87: Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition. Issued February 1987; 82 pp.

This report addresses the questions: what is bioassay? why should it be done? on whom should it be done, and how often? how can the results be interpreted? The only bioassay currently required by regulations in Canadian hospitals is that of radioiodine in the thyroid, so this report is of peripheral interest, unless you want to know more about bioassays in general, and why the AECB thinks it's a good idea to do a least one.

NCRP Report No. 94: Exposure of the Population in the United States and Canada from Natural Background Radiation. 1987

NCRP Report No. 99: Quality Assurance for Diagnostic Imaging. (1989)

NCRP Report No. 102: Medical X-Ray, Electron Beam and Gamma-Ray Protection for Energies Up to 50 MeV (Equipment Design, Performance and Use). (1989)

NCRP Report No. 105: Radiation Protection for Medical and Allied Health Personnel. (1989)

NCRP Report No. 107: Implementation of the Principle of As Low As Reasonably Achievable (ALARA) for Medical and Dental Personnel. (1990)

NCRP Report No. 111: Developing Radiation Emergency Plans for Academic, Medical or Industrial Facilities. (1991)

NCRP Report No. 112: Calibration of Survey Instruments Used in Radiation Protection for the Assessment of Ionizing Radiation Fields and Radioactive Surface Contamination. (1991)

NCRP Report No. 116: Limitation of Exposure to Ionizing Radiation. (1993)

NCRP Report No. 124: Sources and Magnitude of Occupational and Public Exposures from Nuclear Medicine Procedures. 1996.

The report begins with a discussion of the public significance of exposures resulting from nuclear medicine, provides a brief overview of the procedures used in diagnosis and treatment of various conditions, and covers radioimmunoassay techniques. This is followed by a review of the concept of risk from radiation exposures. Included is a brief review of radiation effects with particular emphasis on effects from exposure to low levels of radiation. The subject of radiation exposure to individuals involved in preparing radiopharmaceuticals for administration to patients is followed by a review of exposures of others who may provide care to the nuclear medicine patient or who may be exposed incidentally. Finally, there is a discussion of radiation safety procedures to be observed when providing care to a patient hospitalized after receiving a therapeutic administration of radiopharmaceuticals.

NCRP Report No. 127: Operational Radiation Safety Program. 1998

The report, published June 1998, is an update and revision of NCRP Report No. 59, Operational Radiation Safety Program. The new report has 11 sections, a glossary, and references. The sections are: (1) Introduction, (2) Application of ALARA, (3) Organization and Administration, (4) Facility Design, (5) Orientation and Training, (6) External Radiation Exposure Control, (7) Internal Radiation Exposure Control, (8) Control of Low-Level Radioactive Waste, (9) Control of Exposure to the Public, (10) Radiation Safety Instrumentation, and (11) Planning for Radiation Emergencies. The Report reiterates the basic principles for establishing and maintaining an effective operational radiation safety program. The Report does not try to summarize the regulatory or licensing requirements of the various federal, state, and local authorities that may have jurisdiction over matters discussed.

NCRP Report No. 128: Radionuclide Exposure of the Embryo/Fetus. 1998: 287 pp.

The report is designed to provide information on radiation dose to the embryo/fetus from radionculides in the mother. It has ten sections making up some 92 pages consisting of an introduction, sources of exposure, review of recommendations and regulations regarding exposure of the embryo/fetus, prenatal development, maternal-fetal exchange, prenatal irradiation effects, fetoplacental concentrations and radiation doses, estimation of embryo/fetus dose in radiation protection practice, research needs, and a summary and conclusions. A large part of this Report, 125 pages, provides biological information, fetal/placental information and radiation dose estimates for 83 radionuclides.

NCRP Commentary No. 7: Misadministration of Radioactive Material in Medicine-Scientific Background. 1991.

NCRP Commentary No. 7 is concerned with an assessment of the effects of various levels of radiation exposure that might result from the misadministration of radiopharmaceuticals. Practitioners in the nuclear medicine field and the Nuclear Regulatory Commission have been concerned about proposed reporting requirements for misadministrations. The Commentary provides information on effects that might result from various levels of exposure. Commentary No. 7 is intended to provide a scientific basis for consideration of nuclear medicine misadministrations. Major sections of the Commentary treat the quantity of radioactive materials routinely administered in nuclear medicine procedures, frequency of misadministrations in nuclear medicine and expected radiobiological effects—deterministic and stochastic.

NCRP Commentary No. 9: Considerations Regarding the Unintended Radiation Exposure of the Embryo, Fetus or Nursing Child. 1994.

NCRP Commentary No. 9 seeks to (1) draw special attention to the problems in protection of the embryo, fetus and nursing child that might result from the use of radiation in the medical diagnosis and treatment of the mother, and (2) assist the Nuclear Regulatory Commission in developing requirements appropriate to dealing with the unintended exposure of the embryo, fetus or nursing child as a result of such procedures. The Commentary highlights the fact that physicians must be constantly alert for the patient who may be pregnant or breast feeding. Commentary No. 9 summarizes the doses to the embryo, fetus or nursing child that might result from radiological procedures, brachytherapy to the mother, teletherapy to the mother, and the administration of radiopharmaceuticals to the mother. The Commentary then goes on to treat the risks attributable to these radiation exposures including those for deterministic effects and stochastic effects. Finally, the Commentary sets out recommendations and conclusions aimed at the specification of requirements for action after radiation exposure of the embryo, fetus or nursing child.

NCRP Commentary No. 11: Dose Limits for Individuals Who Receive Exposure from Radionuclide Therapy Patients. 1995.

NCRP Commentary No. 11 is a result of concern about the exposure of family members of recently discharged radionuclide therapy patients. Family members of radionuclide therapy patients would usually be considered members of the public and, therefore, would normally be limited to 1 mSv annually. However, because of the infrequent nature of the exposure and because of the benefits that accrue to a family from a patient's radiation therapy, Commentary No. 11 considers it acceptable for family members of such patients to accrue up to 5 mSv annually and that there may be specific instances when the patient's physician would allow a member of the patient's family to receive an exposure of up to 50 mSv annually. In cases when such an exposure is anticipated, training and individual monitoring should be provided. In the Commentary, there is a discussion of the different types of therapy that can be provided and of the potential exposure from each therapeutic modality. Material in an appendix provides examples of the various situations that can occur.

NCRP Commentary No. 13: An Introduction to Efficacy in Diagnostic Radiology and Nuclear Medicine (Justification of Medical Radiation Exposure). 1995.

NCRP Commentary No. 13 discusses the concept of efficacy as it applies to the use of radiation in diagnostic radiology and nuclear medicine and describes the interrelationships of efficacy with cost-effectiveness and cost-benefit analysis, including consideration of potential radiation detriment, as well as with outcome research and technology assessment. Inherent in the decision to use radiation as a tool in diagnosis is the understanding that every radiation exposure needs to be justified. Since it is only through the development of medical decision-making concepts such as efficacy that decisions concerning justification can be made, the NCRP finds the evaluation of radiation exposure involved in a practice or procedure an essential part of evaluating efficacy. The Commentary presents a general hierarchical model to classify efficacy studies. This hierarchy extends from basic laws of physics, through clinical use, to more general patient outcome and societal issues. Discussion of this hierarchy is followed by a discussion of the applications of efficacy concepts to the assessment and emergence of a new technology. Next, the relations of outcome research, technology assessment, and efficacy are explored. The final section summarizes the logical relationships of efficacy levels and the use of imaging efficacy concepts in clinical decision making.

A7.7 U.N.S.C.E.A.R.

Established in 1955, The United Nations Scientific Committee on the Effects of Atomic Radiation typically consists of about 100 expert delegates representing some 20 countries, plus representatives form such organizations as the International Atomic Energy Agency (I.A.E.A.), the World Health Organization (W.H.O.), the United nations Environment Programme (UNEP), the International Commission on Radiological Protection (ICRP), and the International Commission on Radiation Units and Measurement (ICRU).

Their mandate is: "to review the radiation levels to which the world population is at present, or may in the future become, exposed and the effects and risks that could derive from such exposures." Their annual reports to the U.N. General Assembly are considered among the most authoritative statements on radiation levels and effects, and are often quoted or misquoted in support of arguments for and against nuclear power and other activities involving radiation.

Reports are obtainable from bookstores throughout the world, or from the United Nations, Sales Section, New York.

A7.8 B.E.I.R.

The U.S. National Academy of Sciences has, so far, convened four committees to report on the **Biological Effects of Ionizing Radiation**, which have become known as the BEIR Reports. The first report, known as BEIR I, was released in 1972; the second report, BEIR II, was on cost/benefit analysis for activities involving ionizing radiation and was released in 1977. The third report, "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation" (BEIR III) was released in summary form in May 1979, but withdrawn and released again in November 1980. The BEIR I and BEIR III reports are at the centre of the controversy surrounding the risk of cancer from low levels of radiation, and are much quoted in discussions of this subject. BEIR IV, released in 1988, deals with the health effects of internally deposited alpha emitters, such as radon, and its decay products and is therefore of more interest to the uranium mining community than the hospital community.

Latest publication: BEIR V, Health Effects of Exposure to Low Levels of Ionizing Radiation" 421 pages available from the National Academy Press, Washington, in January, 1990. Revised data for the Hiroshima-Nagasaki survivors resulted in higher values for the estimate of the risk of radiation induced cancer.

A7.9 ASSOCIATIONS

Like most professionals, radiation protection professionals have societies and associations through which they communicate with each other. Most countries have national associations which in turn send delegates to the **International Radiation Protection Association** (IRPA), which holds an international congress every four years.

For Canadians, the most important of these societies is undoubtedly the **Canadian Radiation Protection Association**. The CRPA was founded in 1979 and incorporated in 1982 to foster the development of the science of radiation protection, further the exchange of information, and assist in the development of professional standards. It hosts an annual convention in Canadian locations, publishes a bulletin, and has an energetic medical radiation committee. Various categories of membership are available, depending on the qualifications and experience of the applicant. Application forms may be obtained from:

Secretary, Canadian Radiation Protection Association, P.O. Box 149, Kemptville, Ontario, K0G 1J0, Tel: (613) 230-4883.

Second in importance is possibly the U.S. **Health Physics Society**, which by virtue of its greater membership and resources, makes available a greater wealth of technical information in the form of its monthly journal "Health Physics", and its more casual "HPS Newsletter". Fees are about twice that of the CRPA, but still a bargain, and a variety of membership categories are available. Application forms may be obtained from: The Executive Secretary, the Health Physics Society, 8000 Westpark Drive, Suite 400, Maclean, Virginia 22102, U.S.A.

The **Society for Radiological Protection** (Great Britain), publishes the Journal of Radiological Protection quarterly. Enquiries about membership should be made to the Membership Secretary, Mr. H.C. Orchard, Mount Cottage, Sparrows Green, Wadhurst, E. Sussex, England, TN5 6UH.

The **Campus Radiation Safety Officers** (CRSO) organization is an informal association of RSOs working at educational establishments, and also may be of interest to hospital RSOs, especially those connected with universities. There is no fee to join, and meetings are held in various locations in the US and Canada every two years. It has been suggested that anyone wanting their name on the mailing list should contact the editor of The HPS Newsletter to learn the identity of the current "CRSO Designated Communicator", which changes at frequent intervals. The present address of the HPS Newsletter editor is T. Lew Pitchford, University of Missouri, 518 Clark Hall, Columbia, Missouri, 65211.

Other organizations which can be contacted include the COMP, Canadian Organization of Medical Physics, Box 369, Station K, Toronto, Ontario M4P 2G7; and CAMRT, Canadian Association of Medical Radiation Technologists, 294 Albert Street, Suite 60, Ottawa, Ontario K1P 6E6 and L'ordre des Technologues en Radiologie du Québec, 7400, boulevard Galerie d'Anjou, Bureau 420, Anjou, Québec H1M 3M2.

The NRRPT is a U.S. based organisation.

References

Hy87 The Origins and Work of the International Commission on Radiological Protection, Hylton Smith, M.C. Thorne, Investigative Radiology, Vol 22, No. 11, pp 918-921, November 1987.

APPENDIX XA1
MATHEMATICAL REVIEW

XA1.1 EXPONENTIAL

If we have some number "a" (where a is not = 1), the function f defined by $f(x) = a^x$ is called **the exponential function with base a**.
e.g. $f(2) = a^2$; $f(2/3) = a^{2/3}$; $f(0) = a^0 = 1$ etc.

Some rules for exponents are as follows:
$a^x a^y = a^{x+y}$
$a^x / a^y = a^{x-y}$
$(a^x)^y = a^{xy}$

Some examples with base 2 and base 10 are as follows:

$2^3 \cdot 2^4 = 2^{3+4} = 2^7$	$10^3 \cdot 10^4 = 10^7$
$2^3/2^4 = 2^{3-4} = 2^{-1}$	$10^3/10^4 = 10^{3-4} = 10^{-1}$
$(2^3)^4 = 2^{3x4} = 2^{12}$	$(10^3)^4 = 10^{3x4} = 10^{12}$

Note that the exponent and base need not be integers:

$2^{3.3219} = 10$	$10^{2.3010} = 200$
$2.7183^{2.3026} = 10.000$	

XA1.2 LOGARITHMS

If $y = a^x$, then by definition $\log_a y = x$, where "$\log_a y$" is **the logarithm of y to the base a**.

The logarithm is the inverse of the exponential function.

Some properties and examples:
If $y = 2^3$, then $\log_2 y = 3$
If $y = 10^3$, then $\log_{10} y = 3$
$\log_a x \cdot y = \log_a x + \log_a y$

e.g. $\log_2 (2^3 \cdot 2^4)$	$= \log_2 2^3 + \log_2 2^4$
	$= 3 + 4 = 7$
$\log_{10} (10^3 \cdot 10^4)$	$= \log_{10} 10^3 + \log_{10} 10^4$
	$= 3 + 4 = 7$
$\log_a x/y$	$= \log_a x - \log_a y$
e.g. $\log_2 (2^3/2^4)$	$= \log_2 2^3 - \log_2 2^4$
	$= 3 - 4 = -1$

XA1.3 DIFFERENTIATION

A differential expresses how the change in one quantity depends on the change in another quantity. For example, if we had a reservoir from which water was being pumped, we would notice a change in the volume of water remaining, which would depend on the change in time. In calculus notation, this would be the differential of volume (V) with respect to time (t), written as dV/dt.

To find dV/dt we measure a change in V (i.e. dV) and divide by the corresponding change in time dt.

If the waste is being pumped at a steady rate, dV/dt will always be the same, and will be any volume change divided by the corresponding time change. Thus $dV/dt = (V_2 - V_1)/dt$, where V_1 is the initial volume in the reservoir, V_2 is the final volume, and dt is the time difference between them.

If the flow rate is changing, then dV/dt is changing. To accurately track the change in flow rate, we must measure it over a short time interval, i.e. dt must be small. For high accuracy, dt must be infinitesimally small, i.e. it must approach zero.

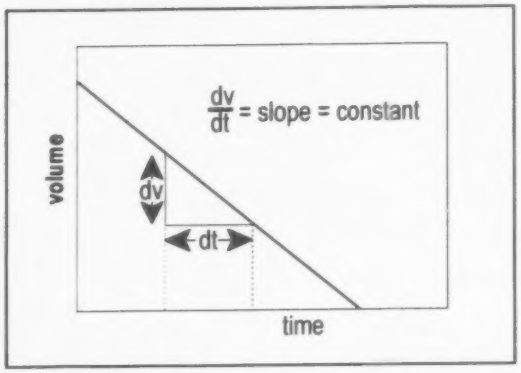
In calculus notation, this situation is written

$$\frac{dV}{dt} = \lim_{dt \to 0} \frac{V(t+dt) - V(t)}{dt}$$

where V(t) is the volume at time t and V(t + dt) the volume at a time dt later.

As we make dt smaller the expression will approach a definite value (if it doesn't, then V is not differentiable with respect to t).

For example, if the flow rate is constant, we can express the volume at any time t by the equation $V(t) = V_0 - a\,t$, where V_0 is the volume when t = 0, and "a" is a constant.



Then

$$\frac{dV}{dt} = \lim_{dt \to 0} \frac{V(t+dt) - V(t)}{dt}$$

$$\frac{dV}{dt} = \lim_{dt \to 0} \frac{\{V_0 - a(t+dt)\} - \{V_0 - at\}}{dt}$$

$$= \lim_{dt \to 0} \frac{V_0 - at - adt - V_0 + at}{dt}$$

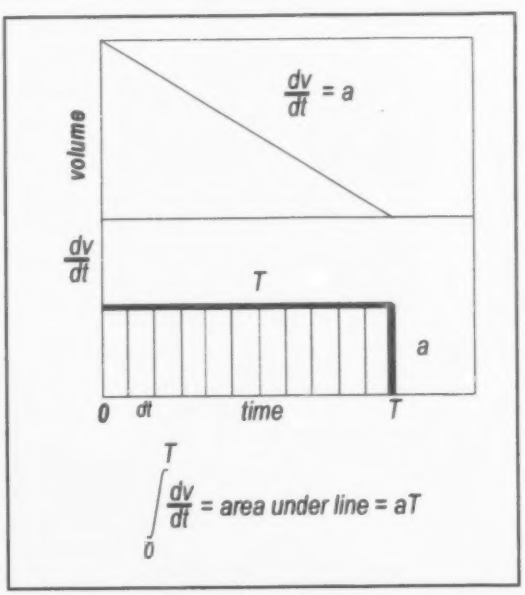
$$= \lim_{dt \to 0} - \frac{adt}{dt} = \lim_{dt \to 0} - a = - a$$

Here, a is the flow rate (e.g. litres per second). It comes out negative because the volume in the reservoir is decreasing as time is increasing. If we substitute the number of radioactive atoms remaining (N) for volume of water remaining (V) in the above example, then the differential of N with respect to t is dN/dt, which is the activity of the source. However, as we shall see, dN/dt is not constant over time, as dV/dt was above.

XA1.4 INTEGRATION

Integration is the inverse of differentiation. To use the same water reservoir example as before, if we collected (integrated) the water flow dV/dt between the initial and final time limits, we would get the whole volume of water that was in the reservoir.

Just as differentiation was the change in volume in an infinitesimal time period, integration is the sum of all the volume changes for all the infinitesimal



time periods. The symbol for integration is an elongated "s" for sum i.e. "$\int$".

The integration of the flow dV/dt from time t = 0, to some later time T is written as

$$\int_0^T \frac{dV}{dt}\,dt = \Big[V\Big]_0^T = \left[V_0 - V_T\right]$$

If T was just the time at which the reservoir ran out, then the remaining volume at time T (i.e. V_T) would be 0, and the integral of the flow from t = 0 to t = T would be V_0.

If, as before, the volume at any time t is given by $V(t) = V_0 - a\,t$, then dV/dt = -a and

$$\int_0^T \frac{dV}{dt}dt = \int_0^T -a\,dt = \Big[-at\Big]_0^T = \left[aT\right]$$

Rules for integrating various functions can be found in most calculus text books.

XA1.5 Exponential "e"

We previously found the differential of the function $V = V_0 - at$. Suppose instead the function is $V = a^t$. The differential dV/dt is

$$\frac{dV}{dt} = \lim_{dt\to 0} \frac{V(t+dt) - V(t)}{dt}$$

$$= \lim_{dt\to 0} \frac{a^{t+dt} - a^t}{dt}$$

$$= \lim_{dt\to 0} \frac{a^t(a^{dt} - 1)}{dt}$$

$$= a^t \lim_{dt\to 0} \frac{(a^{dt} - 1)}{dt}$$

Note that the limit does not involve t, i.e. it is some constant k. Thus we can say that $dV/dt = k\,a^t$.

We now ask the question: is there some value of the base "a" for which the value of k is 1? The answer is "yes". When the base a is equal to 2.718281828459045... This irrational number is denoted by the letter "e", after the Swiss mathematician Leonhard Euler.

Thus, if $V = e^t$, $dV/dt = e^t$,
i.e. the differential of e^t is e^t.

There are other ways of defining the "exponential e", but this one best suits our purpose.

XA1.6 Natural Logarithms

We previously defined the logarithm as follows: if $y = a^x$, then $\log_a y = x$. When the base a is chosen as e, then we call the logarithm the "natural logarithm", usually written as ln y.

Thus, if $y = e^x$, $\ln y = x$;
or, if $V = e^t$, $\ln V = t$.

Note that since exponentiation is the inverse of taking the logarithm, when the two operations are performed in sequence, they "cancel out". Thus, for example, $10^{\log 100} = 100$ (using 10 as base) and $e^{\ln t} = t$ (using e as base).

Starting with $e^{\ln t} = t$, we can find the differential of ln t with respect to t. We will let $u = \ln t$; then $t = e^u$. Differentiate both sides of this equation with respect to t:

$$\frac{d}{dt}(t) = \frac{d}{dt}(e^u)$$

$$i.e.\ 1 = \frac{d}{du}(e^u) \times \frac{du}{dt}$$

where we have used the "chain rule" which says that

$$\frac{dv}{dt} = \frac{dv}{du} \times \frac{du}{dt}$$

Now, $\frac{d}{du}(e^u) = e^u$

and $\frac{du}{dt} = \frac{d}{dt}(\ln t)$

So we have $1 = e^u \cdot \frac{d}{dt}(\ln t)$

i.e. $\frac{d}{dt}(\ln t) = \frac{1}{e^u} = \frac{1}{t}$

Integrating both sides of this equation:

i.e. $\int \frac{d}{dt}(\ln t) = \int \frac{1}{t}$ i.e. $\int \frac{1}{t} = \ln t$

In general then $\int \frac{1}{x} dx = \ln x$

XA1.7 Radioactive Decay

We can now apply the above calculus to various problems related to radioactivity and radiation, starting with the simple decay of a radioactive source.

The fundamental law of radioactive decay is that the number of transformations occurring per unit time (e.g. per second) is proportional to the number of unstable atoms available for transformation.

In calculus notation, $dN/dt = -\lambda N$, where N is the number of unstable atoms at time t, and λ is the constant of proportionality (the decay constant). Rearranging this equation we have:

$$\frac{1}{N} dN = -\lambda dt$$

Integrating both sides of this equation:

$$\int \frac{1}{N} dN = -\lambda \int dt$$

i.e. $\ln N = -\lambda t$

Here we have integrated without limits on t, i.e. we have an "indefinite integral". We therefore add a "constant of integration" C.

$$\ln N = -\lambda t + C$$

But the equation has a "boundary condition" which must be obeyed, which is: when the time t=0, the number of unstable atoms N is N_0.

Putting $t = 0$ and $N = N_0$ gives: $\ln N_0 = C$

So the full equation is $\ln N = -\lambda t + \ln N_0$

i.e. $\ln N - \ln N_0 = -\lambda t$
i.e. $\ln (N/N_0) = -\lambda t$
i.e. $N/N_0 = e^{-\lambda t}$
i.e. $N = N_0 e^{-\lambda t}$,

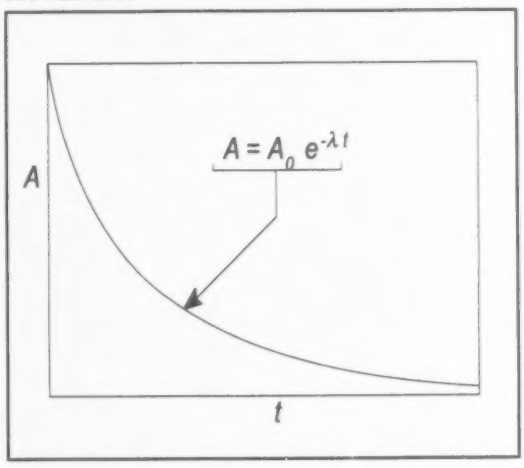
which shows how the number of remaining unstable nuclei N decreases with time.

But the activity A is $dN/dt = -\lambda N$, and the initial activity $A_0 = -\lambda_0$

So $N = -A/\lambda$ and $N_0 = -A_0/\lambda$.

Substituting these values in the equation above gives $A = A_0 e^{-\lambda t}$, which shows how the activity A decreases with time.

The half life, $T_{1/2}$, is the time required to reduce the number of nuclei to half the original value. Since activity is directly proportional to the number of nuclei, the activity is also reduced to half after one half life.



Putting $A = A_0/2$, and $t = T_{1/2}$ in the equation $A = A_0 e^{-\lambda t}$ gives

$A_0/2 = A_0 e^{-\lambda T_{1/2}}$

Taking ln of both sides gives $\ln 1/2 = -\lambda T_{1/2}$

i.e. $\ln 1 - \ln 2 = -\lambda T_{1/2}$

$\ln 1 = 0$, so $T_{1/2} = \ln 2/\lambda$

The mean life (τ) is simply the average time an unstable nucleus exists before undergoing transformation. Suppose we started with a total of 12 nuclei, which had lifetimes as follows: 6 nuclei lasted 1 second, 3 nuclei lasted 2 seconds, 2 nuclei lasted 3 seconds, and 1 nucleus lasted 4 seconds. Then the average or mean lifetime of all the nuclei would be $(6 \times 1 + 3 \times 2 + 2 \times 3 + 1 \times 4)/12 = 22/12 = 1.83$ seconds.

Note that we multiplied each lifetime by the number of nuclei having that lifetime (e.g. 3 nuclei x 2 seconds), summed over all lifetimes, and divided by the total number of nuclei. Let us repeat the same operation for N_0 nuclei (instead of 12), and use a small time interval dt instead of the 1 second interval used above.

At some time t, the number of remaining nuclei is $N = N_0 e^{-\lambda t}$. During the next dt seconds, a number of dN nuclei will transform, where $dN = -\lambda N dt$ (since $dN/dt = -\lambda N$). These dN nuclei had a lifetime = t.

So, multiplying the number of atoms by their lifetime value gives $dN \times t$. Now we repeat this for all lifetime values and add it all up, i.e. Sum ($dN \times t$) for all values of t = Sum ($\lambda N dt \times t$) for all t from 0 to ∞.

In calculus notation this is

$$\int_0^\infty \lambda N t\,dt = \int_0^\infty \lambda t N_0 e^{-\lambda t} dt$$

$$= \lambda N_0 \int_0^\infty t e^{-\lambda t} dt$$

Now $\int_0^\infty t e^{-\lambda t} dt = \left[\frac{1}{\lambda^2}(\lambda t - 1) e^{-\lambda t}\right]_0^\infty$

and $= \frac{1}{\lambda^2}$ [since $e^{-\infty} = 0$ and $e^0 = 1$]

So, multiplying by the λN_0 gives $\frac{N_0}{\lambda}$ for the

total sum (the 22 in our previous simple example). Now divide by the total number of nuclei N_0 and

we get mean life $\tau = \frac{1}{\lambda}$

$= \frac{T_{1/2}}{\ln 2}$ (since $\lambda = \ln 2/T_{1/2}$) $= 1.44\ T_{1/2}$

In section XA1.4 above, we noted that if we integrated (collected) all the water flow dv/dt from time zero until the reservoir was empty, we should get the initial volume in the reservoir V_0. Similarly, if we integrate the activity dN/dt over all time, we should get the original number of nuclei N_0. The integral of activity from 0 to infinite time is

$$\int_0^\infty A\,dt = \int_0^\infty A_0 e^{-\lambda t} dt = A_0 \int_0^\infty e^{-\lambda t} dt$$

$$= A_0 \left[\frac{1}{-\lambda} e^{-\lambda t}\right]_0^\infty = -A_0 \frac{1}{\lambda}$$

$$= -A_0 \tau = -A_0\, 1.44\, T_{1/2}$$

We can show that this is, in fact equal to N_0 as follows:

$A = dN/dt = -\lambda N$, so $-A/\lambda = N$
and $-A_0/\lambda = N_0$.

XA1.8 Photon Attenuation

The mathematical treatment of the attenuation of photons (X or gamma rays) by a shielding material (e.g. lead) is identical to the above, with the number of photons (or intensity of radiation I) replacing the number of nuclei N, and distance into the material x replacing the time t.

Thus, starting from the premise that the number of photons removed from the beam at any point (dI) is proportional to the number of photons at that point (I) we have $dI/dx = -\mu x$. Here the derivative is with respect to distance x, and the linear attenuation coefficient μ replaces the decay constant λ. Following the same route as before we get $I = I_0 e^{-\mu x}$ where I is the radiation intensity at depth x into the material, and I_0 is the original intensity (at depth 0).

The depth at which the intensity is reduced to half the original value is the half value layer (HVL), and $x_{1/2} = \ln 2/\mu$.

XA1.9 Parent/Daughter Decay

Suppose we have a radionuclide which decays to a second (daughter) radionuclide, which in turn decays to a third nuclide. How does the activity of the daughter radionuclide vary with time?

Let the number of parent nuclei be M at any time t, and the number of daughter nuclei be N. Let the parent and daughter decay constants be λ_M and λ_N respectively.

The rate of transformation of parent nuclei is
$dM/dt = \lambda_M M$

The rate at which daughter nuclei are created is the rate at which parent nuclei transform; i.e. $\lambda_M M$. But, as they are created, daughter nuclei are transforming at a rate of $\lambda_N N$. So the <u>net</u> rate of change in the number of daughter nuclei is

$dN/dt = \lambda_M M - \lambda_N N$.

But $M = M_0 e^{-\lambda_M t}$

So $dN/dt = \lambda_M M_0 e^{-\lambda_M t} - \lambda_N N$

i.e. $\frac{dN}{dt} + \lambda_N N = \lambda_M M_0 e^{-\lambda_M t}$

Multiplying both sides of this equation by $e^{(\lambda_N t)}$

$$e^{\lambda_N t}\frac{dN}{dt} + \lambda_N N e^{\lambda_N t} = \lambda_M M_0 e^{(\lambda_N - \lambda_M)t}$$

$$e^{\lambda_N t}\frac{dN}{dt} + \lambda_N N e^{\lambda_N t} = \lambda_M M_0 e^{(\lambda_N - \lambda_M)t}$$

ie. $\frac{d}{dt}(Ne^{-\lambda_N t}) = \lambda_M M_0 e^{(\lambda_N - \lambda_M)t}$

Integrating both sides:

$$Ne^{-\lambda_N t} = \frac{\lambda_M}{\lambda_N - \lambda_M} M_0 e^{(\lambda_N - \lambda_M)t} + C$$

(C = constant of integration)

To find C, we assume that N = 0 at t = 0
(no daughter initially)

Then $0 = \frac{\lambda_M}{\lambda_N - \lambda_M} M_0 + C$

i.e. $C = -\frac{\lambda_M}{\lambda_N - \lambda_M} M_0$

And the full equation is:

$$Ne^{\lambda_N t} = \frac{\lambda_M}{\lambda_N - \lambda_M} M_0 e^{(\lambda_N - \lambda_M)t} - \frac{\lambda_M}{\lambda_N - \lambda_M} M_0$$

i.e. $$N = \frac{\lambda_M}{\lambda_N - \lambda_M} M_0 (e^{-\lambda_M t} - e^{-\lambda_N t})$$

Which shows how the number of daughter nuclei, N, varies with time.

The activity of the daughter $A_N = \lambda_N N$

and the initial activity of the parent

$A_0 = \lambda_M M_0$. so $M_0 = A_0 / \lambda_M$

Thus $$A_N = A_0 \frac{\lambda_{NM}}{\lambda_N - \lambda_M} (e^{-\lambda_M t} - e^{-\lambda_N t})$$

Example:

Half life of parent = 66 h; $\lambda_M = 1n\ 2/66 = 0.010$

Half life of daughter = 6 h:

$\lambda_N = 1n\ 2/6 = 0.1155$ h-1

Initial activity of parent = 10 Gbq

Then, activity of daughter at t = 10 h is

$$10\ Gbq \cdot \frac{0.1155}{0.1155 - 0.0105} \cdot (e^{-0.0105} - e^{-1.155}$$

$$= 1.97\ \text{GBq}$$

At t = 20 h, the daughter activity is 7.82 Gbq

At t = 30 h, the daughter activity is 7.68 GBq

And so on.

(If this is Mo-99/Tc-99m, these results would have to be multiplied by 0.86, since only 86% of Mo-99 decays lead to Tc-99m).

It can be shown that the daughter activity reached a maximum at time

$$t_{max} = \frac{1n\ \lambda_M / \lambda_N}{\lambda_M - \lambda_N} = 22.84\ \text{h.}$$

for the above example.

It can also be shown that the parent and daughter activities are equal at this time.

APPENDIX XA2
SAMPLE CALCULATIONS

These examples are included mainly to illustrate and reinforce the meaning of the various radiation quantities and units. Some of them will rarely or never have to be performed by a hospital Radiation Safety Officer.

XA2.1 ACTIVITY

Example 2.1.1. A technologist is instructed to administer 750 MBq of Tc-99m to a patient, but the (old) dose calibrator reads in curies. How many Ci should be drawn up?

Solution:

37×10^9 dps = 1 Ci
37×10^6 dps = 1 mCi
750 MBq = 750×10^6 dps = 750/37 mCi
= 20.3 mCi

Example 2.1.2: A standard source of Co-57 is to be used to check the calibration of a dose calibrator. The assay information on the source, imported from the U.S., is 5735 microcuries on May 1, 1987. Your new dose calibrator assays the source as 36.2 MBq on 17 March 1989. Is the dose calibrator working properly?

Solution:

The half life of Co-57 is 270.9 days
Elapsed time from May 1, 1987 to March 17, 1989
= 687 days

$A = A_0 e^{-\lambda t}$ where $\lambda = \ln 2/T$ where T is the half life.

i.e. $A = A_0 e^{-\ln 2\, t/T} = 5735\ \mu Ci \times e^{-\ln 2 \times 687/270.9}$

A = 989 μCi = 36.6 MBq on March 17
Dose calibrator reading = 36.2 MBq
(i.e. within 1% of true)

Example 2.1.3: An I-131 therapy capsule is assayed at 3.6 GBq. How long must the capsule be stored to reduce its activity to 500 kBq?

Solution:

Half life of I-131 = 8.04 days
3.6 GBq = 3.6×10^{12} Bq; 500 kBq = 5×10^5 Bq
Using the equation $A = A_0 e^{-\ln 2\, t/T}$ we have
5×10^5 Bq = 3.6×10^{12} Bq $\times e^{-\ln 2\, t/8.04}$ where t is the decay time
i.e. $1.389 \times 10^{-7} = e^{-\ln 2\, t/8.04}$
Taking the natural log of both sides of the equation gives
$\ln (1.389 \times 10^{-7}) = -\ln 2\ t/8.04$
i.e. $t = -15.79 \times 8.04/-\ln 2 = 182$ days

Example 2.1.4: A 4.0 GBq therapy dose of I-131 is administered to a patient at 10 a.m. on Monday. Assuming 75% of this is excreted in the urine during the next 24 hours, what activity remains in the patient at 10 a.m. on Saturday?

Solution:

Neglecting physical decay during the first 24 hours, assume 25% × 4 GBq = 1 GBq remains at 10.am. on Tuesday
Elapsed time to 10 a.m. Saturday = 4 days
Therefore A = 1 GBq × $e^{-\ln 2 \cdot 4 \text{ days}/8 \text{ days}}$
= 0.707 GBq = 707 MBq

Example 2.1.5: A Mo-99/Tc-99m generator is eluted at 8.00 a.m. and yields 14.6 GBq. Certain quantities are used at various times during the day as shown in the log below. What activity remains at 5.00 p.m. on the same day?

Time	**Activity Withdrawn**
8:45 a.m.	750 MBq
9:45 a.m.	400 MBq
11:30 a.m.	1500 MBq
2:45 p.m.	750 MBq

Solution:

Half life = 6 h

8:00 a.m.		14.600 GBq
8:45 a.m.	14.6 × $e^{-\ln 2 \cdot 0.75 h / 6 h}$ =	13.388 GBq
	withdraw	0.750 GBq
		12.638 GBq
9:45 a.m.	12.638 × $e^{-\ln 2 \cdot 1.0 h / 6 h}$ =	11.259 GBq
	withdraw	0.400 GBq
		10.859 GBq
11:30 a.m.	10.859 × $e^{-\ln 2 \cdot 1.75 h / 6 h}$ =	8.872 GBq
	withdraw	1.500 GBq
		7.372 GBq
2:45 p.m.	7.372 × $e^{-\ln 2 \cdot 3.25 h / 6 h}$ =	5.064 GBq
	withdraw	0.750 GBq
		4.314 GBq
5:00 p.m.	4.314 × $e^{\ln 2 \cdot 2.25 h / 6 h}$ =	3.327 GBq

Example 2.1.6: A contamination monitor has detection efficiency for various radionuclides as shown in the table. When the probe is placed on a benchtop surface, a net count rate of 3000 cpm is obtained. If the contamination is Tc-99m, what is the contamination level in Bq/cm^2?

If the contamination is I-131, what is the contamination level?

Radionuclide	**cpm per Bq/cm^2**
C-14	300
Tc-99m	200
I-125	200
I-131	1500

Solution:

Tc-99m:
3000 cpm÷ 200 cpm·cm^2/Bq = 15 Bq/cm^2

I-131 :
3000 cpm ÷ 1500 cpm·cm^2/Bq = 2.0 Bq/cm^2

The detection efficiency table is adapted from that published for the Berthold 1210B contamination monitor. Other contamination monitors would have different efficiency tables.

Note the factor of 7.5 difference in results depending on which radionuclide is causing the contamination. Where the contaminating radionuclide is unknown, it is impossible to accurately determine the level of contamination, and a worst case should be assumed.

Example 2.1.7: A 100 cm^2 area of a workbench surface is swipe tested, and the swipe counted on a thyroid uptake probe to give 240 cpm. The background count is 120 cpm. The probe calibration is given. Assuming the swipe removed 10% of the surface contamination, what is the surface contamination level in Bq/cm^2 if the contamination is Tc-99m, and if the contamination is I-131?

Probe Calibration	cpm/Bq
Tc-99m	1.2
I-131	3.6
I-125	2.2

Solution:

Net cpm = 240 - 120 = 120 cpm

Tc-99m:
activity = 120 cpm ÷ 1.2 cpm/Bq = 100 Bq
This is 10% of total contamination.
Therefore total = 1000 Bq
This 1000 Bq is assumed to be uniformly spread over the 100 cm^2 wiped
Therefore contamination level = 10 Bq/cm^2

(b) I-131:
activity = 120 cpm ÷ 3.6 cpm/Bq = 33 Bq
Therefore contamination level = 3.3 Bq/cm^2

Note: A thyroid probe, due to lower counting efficiency and poorer shielding against background, is not as sensitive as a well counter for measuring wipes.

XA2.2 EXPOSURE

Example 2.2.1: The radiation level at a workbench due to nearby radionuclide storage is measured at a constant 25 μGy/h. If a technologist spends 8 hours per day at this location, what is her accumulated absorbed dose for a 5 day week? Assuming she works 50 weeks per year, what is her annual equivalent dose?

(Note: if an older exposure rate meter was used, the reading would be 2.5 mR/h. Here we are assuming the manufacturer has changed the monitor scale to read in μGy/h, as discussed in section A4.11).

Solution:

25 μGy/h × 8 h/day = 200 μGy/day
200 μGy/d × 5 d/wk × 50 wk/y = 50,000 μGy
= 50 mGy = 0.05 Gy

Assuming the radiation is gamma, the Quality Factor = 1, and 0.05 Gy absorbed dose = 0.05 Sv equivalent dose (50 mSv)

Example 2.2.2: What is the exposure rate 1 m from a 4 GBq I-131 capsule?
What is the rate at 2 m? At 10 cm? At 1 cm?

Solution:

The exposure rate constant for I-131 is 0.22 mR/h per mCi at 1 m
This may be translated into S.I. units as follows:

1 R = 2.58×10^{-4} C/kg
So 0.22 mR/h = 0.616×10^{-7} C/kg·h
1 mCi = 37×10^{6} Bq
So exposure rate constant 0.616×10^{-7} C/kg·h ÷ 37×10^{6} Bq = 1.66×10^{-15} C/kg·h per Bq at 1 m

4 GBq = 4×10^{9} Bq
So exposure rate 1 m from 4 GBq = 4×10^{9} Bq × 1.66×10^{-15} C/kg·h per Bq = 6.64×10^{-6} C/kg·h

At 2 m, (2 times the distance) the exposure rate will be reduced by a factor of 4
i.e. $6.64 \times 10^{-6} \times 1/4 = 1.66 \times 10^{-6}$ C/kg·h

At 10 cm (one tenth the original distance), it is increased by × 100
i.e. $6.64 \times 10^{-6} \times 100 = 6.64 \times 10^{-4}$ C/kg·h

At 1 cm (one hundredth the original distance), it is increased by × 10000
i.e. $6.64 \times 10^{-6} \times 10000 = 6.64 \times 10^{-2}$ C/kg·h

Example 2.2.3: The dose rate 1 m from a Tc-99m source is measured as 21 µGy/h at 8:00 a.m. What is the total accumulated dose to a person who sits at this location from 8:00 a.m. to 6:00 p.m.?

Solution:

The dose rate D' will decrease at the same rate as the Tc-99m activity
i.e. $D' = D_0' e^{-\lambda t}$

The total dose is the integral of D' from time 0 to some time T. i.e. $D = \int D_0' e^{-\lambda t} dt$ between 0 and $T = (D_0'/\lambda)(1 - e^{-\lambda T})$

(See Appendix XA1.7 for evaluation of the above integral)

$\lambda = \ln 2/6h$, $D_0' = 21$ μGy/h and T = 10 h (from 8 a.m. to 6 p.m.) so $D = (21 \times 6/\ln 2)(1 - e^{-\ln 2 \cdot 10/6}) =$ 124.5μGy

Example 2.2.4: A patient is given 7.5 GBq of I-131. Immediately after the administration, the dose rate 1 m from the patient is 220 μGy/h. The patient is discharged 3 days later, at which time the measured dose rate at 1 m is 32 μGy/h. Estimate the total absorbed dose to a spouse who is assumed to spend 12 h per day thereafter at a distance of 1 m from the patient.

(Note: the exposure rate constant for I-131, in old units, is 22 mR/h per 100 mCi, which predicts a dose rate at 1 m for 7.5 GBq (200 mCi) of 440 μGy/h. The actual measurement on a patient will often be much less, due to tissue attenuation.

Note also that the reduction in dose rate from 220 μGy/h to 32 μGy/h in 3 days is due mostly to an initial, rapid biological excretion into the urine, which is impossible to calculate accurately. Hence, the dose rate at 1 m from the patient is measured prior to release, and this final measurement used to estimate doses subsequent to release.)

Solution:

The physical half life of I-131 is 8 days, and the biological half life for excretion after uptake by the thyroid is about 60 days, giving an effective (combined) half life of about 7 days.

The initial dose rate at 1 m is 32 μGy/h, which will diminish exponentially with a half life of 7 days. To find the total dose for total decay we integrate the dose rate D' as in example 2.2.3 above, but this time from 0 to infinite time.
i.e. $\int D_0' e^{-\lambda t} dt$ from 0 to ∞.

The result of this integration is $1.443 \times D_0' \times T_E$ *(See Appendix XA1.7)*

i.e. Dose = 1.443 × 32 μGy/h × 7 × 24 h
= 7.8 mGy

Finally, since the spouse is irradiated for only half the total time (12 h per day), we divide by 2 for a final dose of 3.9 mGy.

XA2.3 SHIELDING

Example 2.3.1: The Exposure rate 0.5 m from Tc-99m source is 3.2 μC/kg·h (12.4 mR/h). What thickness of lead shielding is required to reduce this to 65 μC/kg·h (0.25 mR/h)?

Solution 1: ***(quick and dirty method)***

Half Value Layer (HVL) for lead for 140 keV (Tc-99m) gamma rays is 0.256 mm
From 3.2×10^{-6} C/kg·h down to 65×10^{-9} C/kg·h is an attenuation factor of about 50.
1 HVL attenuates by a factor of 2
5 HVL's by a factor of $2^5 = 32$
6 HVL's by a factor of $2^6 = 64$
So we need about 5 1/2 HVL's for a factor of 50 attenuation 5.5 × 0.256 mm = 1.4 mm

Solution 2 ***(slow and accurate method)***

Attenuation equation is $I = I_0 e^{-\mu x}$, where
μ = linear attenuation coefficient
= ln2/HVL, and x = thickness of shielding
$\mu = \ln 2/0.256 = 2.71$ mm^{-1}
Then: $(65 \times 10^{-9}$ C/kg·h
$= (3.2 \times 10^{-6}$ C/kg·h
i.e. $2.03 \times 10^{-2} = e^{-2.71x}$
i.e. $\ln (2.03 \times 10^{-2}) = 2.7\ x$
i.e. $-3.9 = -2.7\ x$
So x = 1.44 mm

Example 2.3.2: The Exposure rate 1 m from an I-131 therapy capsule is 5.7 μC/kg·h (22 mR/h). What is the exposure rate at the same position if the capsule is shielded by 2 inches of lead?

Solution:

HVL for lead for I-131 gamma rays is 0.219 cm.
2 inches = 2 × 2.54 = 5.1 cm
5.1 cm = 5.1/0.219 = 23.2 HVL's
Attenuation factor = $2^{23.2}$ = 9,635,980
= 10^7 approximately
5.7×10^{-6} C/kg·h × 10^{-7} = 5.7×10^{-13} C/kg·h
= 0.57 C/kg·h (2.2 nR/h)

Example 2.3.3: A technologist sitting 0.5 m from an unshielded I-131 capsule receives an accumulated exposure of 91.2 μC/kg (353 mR) in 4 hours. If she places 1 cm of lead shielding around the capsule and moves back to a distance of 1.5 m, what will be her accumulated exposure in 2 hours?

Solution:

91.2 μC/kg in 4 hours = exposure rate of 22.8 μC/kg·h. 1cm lead = 4.57 HVL's (1 HVL = 0.219 cm)
So attenuation factor = $2^{4.57}$ = 23.7
So shielded exposure rate = 22.8/23.7 = 0.96 μC/kg·h at 0.5 m
Increase distance to 1.5 m (factor of 3) reduces radiation by factor of 9
New exposure rate = 0.96/9 = 0.107 μC/kg·h
Exposure accumulated in 2 hours
= 0.107 μC/kg·h × 2h = 0.2 μC/kg

Note: exposure was reduced using the classical triad of time (factor of 2 reduction), distance (factor of 9) and shielding (factor of 24).

Total reduction factor = 2 × 9 × 24 = 432
91.2 μC/kg × 432 = 0.2 μC/kg

XA2.4 Absorbed Dose

Example 2.4.1: A 200 g block of plastic is uniformly irradiated by gamma rays and absorbs 12.5 joules of energy. Calculate the average absorbed dose to the plastic in grays and rads.

Solution:

1 gray = an energy absorption of 1 joule per kg.
Here we have 12.5 J per 0.2 kg
= 12.5/0.2 = 62.5 Gy

Example 2.4.2: The same block of plastic is irradiated by neutrons and absorbs the same 12.5 J of energy. Calculate the average absorbed dose again.

Solution:

Energy absorption is the same as before
= 62.5 J.kg
So, absorbed dose is the same = 62.5 Gy

(Absorbed dose depends only on the energy absorbed, not on the type of radiation)

Example 2.4.3: The Exposure rate 1 m from a radioactive source is 258 μC/kg·h (1 röntgen/h). Calculate the absorbed dose in grays and rads to a mass of air at this position for 1 hour.

Solution:

Each negative ion (electron) produced in air has a charge of 1.6×10^{-19} C.
So, to produce 2.58×10^{-4} C of negative charge we need to produce $2.58 \times 10^{-4}/1.6 \times 10^{-19}$ electrons in the 1 kg of air.
It takes 33.85 eV of energy to ionize one air molecule, which is $33.85 \times 1.6 \times 10^{-19}$ joules.

(if you think about the definition of the volt and the electron-volt, you will realize that the reappearance of the electron charge as the conversion factor is no coincidence.)

Therefore, the energy needed to produce the required charge is (2.58×10^{-4} / 1.6×10^{-19} electrons) × $33.85 \times 1.6 \times 10^{-19}$ joules per electron. = 0.00873 joules in 1 kg of air. = 8.73 mGy
(= 87.3 ergs per gram of air = 0.873 rads)

Example 2.4.4: In the above example, the air is replaced by a mass of tissue-equivalent material. Calculate the absorbed dose.

Solution:

In the example above we calculated that 0.00873 joules of energy were deposited in 1 kg of air by the radiation. To calculate what energy would be deposited in tissue, we multiply by the ratio of the energy absorption coefficients for tissue and air. The values of these coefficients vary slightly with energy, but between 100 and 500 kev, their ratio is close to 1.1. Thus, energy absorbed by 1 kg tissue is 0.00873 × 1.1 = 0.0096 joules/kg = 9.6 mGy = 96 ergs/g = 0.96 rads.

It was because an exposure of 1 röntgen results in an absorbed dose to soft tissue of 96 ergs per gram that the rad unit of absorbed dose was defined as an energy absorption of 100 ergs/g. To an approximation sufficiently accurate for radiation protection purposes, an exposure measured at 1 röntgen meant an absorbed dose of 1 rad. For radiations for which the quality factor was unity, 1 rad was also equal to 1 rem. So convenient was this on-to-one correspondence between röntgens, rads and rems that their differences were often ignored, and one could talk of so many "R's" of radiation without being too specific as to what the "R" stood for.

While the S.I. units, the gray and the sievert, retain this one-to-one correspondence for Q=1 radiations, the S.I. unit for exposure, the coulomb/kg has no such simple correspondence.

Example 2.4.5: The air kerma rate constant for I-131 is 52 μGy/h·GBq at 1 m. Calculate the air kerma rate at a distance of 0.5 m from a 500 MBq source of I-131. Calculate the dose rate (in Gy/h) to a volume of air at this point. Calculate the dose rate to a volume of tissue at this point.

Solution:

Recall that kerma is the kinetic energy released in a unit mass of material by an uncharged ionizing particle (e.g. a gamma ray). Absorbed dose equals the kerma less the energy carried away by bremsstrahlung. For gamma ray energies common in hospitals, this bremsstrahlung energy is less than 1%, and can be neglected;
i.e. air kerma = absorbed dose to air.

500 MBq is half of 1 GBq, so the air kerma rate at 1 m would be 26 µGy/h Going to 0.5 m decreases the distance by 2, and therefore increases the radiation by × 4;
i.e. to 104 µGy/h

To calculate the dose rate to tissue we multiply by the ratio of the energy absorption coefficients for tissue and air (1.1) to give 114 µGy/h.

Note that this problem is similar to 3.4.3 and 3.4.4 above, but since air kerma is already expressed in energy absorption units, there is no need to convert from charge/kg (as produced by exposure) to joules/kg.

XA2.5 EFFECTIVE DOSE

Example 2.5.1: A collimated beam of x-rays strikes a patient's lungs, delivering an absorbed dose to them of 0.5 centigray. Calculate the effective dose to the patient.

(Note: a "centigray" (one hundredth of a Gy) is equal to one rad (old units) and sometimes used by RSO's trying to switch from old to S.I. units).

Solution:

Effective Dose $E = \sum W_T H_T$, i.e. the sum over all affected tissues (organs) of the tissue weighting

factor W_T times the dose equivalent to that organ H_T. In this case, there is only one irradiated organ, the lungs.

Dose to lungs = 0.5 centiGy = 5 mGy
W_R for x-rays = 1
So Equivalent Dose H_T = 5 mGy × 1 = 5 mSv
W_T for lung = 0.12 (ICRP 60)
So, effective dose E_E = 0.12 × 5 mSv = 0.6 mSv

(Note: only part of the body received 5 mSv, so the detriment is as if the whole body received 0.6 mSv)

Example 2.5.2: Due to an intake of radioactive material an employee received the following equivalent dose to various organs:
Gonads: 3.2 mSv; Thyroid: 0.6 mSv: Liver: 6.4 mSv. Calculate the effective dose to the employee.

Solution:

Tissue weighting factors are: gonads = 0.2, thyroid = 0.05, liver = 0.05. So, effective dose equivalent is $E = \sum W_T H_T$
i.e. H_E = 0.2 × 3.2 mSv (gonads) + 0.05 × 0.6 mSv (thyroid) + 0.05 × 6.4 mSv (liver) = 0.99 mSv

XA2.6 COMMITTED DOSE

EXAMPLE 2.6.1: A technologist has an acute intake of 0.5 MBq of I-131. What is the committed effective dose from this intake?

Solution:

Various published tables list the dose to the thyroid (of Reference Man) for total decay of unit intake of I-131, e.g. ICRP Publication 52 gives the dose to the thyroid for 35% uptake as 500 mGy/MBq administered so 0.5 MBq would give 250 mGy. Since the emissions are beta and gamma, this is an equivalent dose of 250 mSv. Since the effective half life of I-131 is only 7.5 days, the dose given above for total decay will occur well within the 50 year span over which the dose commitment is to be calculated. So the committed equivalent dose is 250 mSv, and the committed effective dose is 250 mSv × 0.03 = 7.5 mSv, where 0.03 is the tissue weighting factor for thyroid. If the % uptake is known to be different from 35%, the results can be pro-rated accordingly.

XA2.7 ALI, DIL

Example 2.7.1: A radiopharmacist inhales 200 kBq of I-131 during an iodination. How many ALI's is this? If 10% of the intake was later measured in her thyroid, would the DIL for I-131 be exceeded?

Solution:

The Annual Limit of Intake (ALI) is defined as that intake which will produce a committed equivalent dose equal to the annual limit. Calculation of ALI's is based on Reference Man, and for I-131 is 2 MBq. Therefore, 200 kBq represents 200/2000 = 0.1 ALI's.

If the uptake is 10% of the 200 kBq intake, it is 20 kBq. The Derived Investigation Level (DIL) is that uptake which will produce a dose commitment of 1/20 of the annual limit, and for I-131 is 10 kBq (for an ARW Atomic Radiation Worker). Thus, this uptake is twice the investigation level.

XA2.8 INTERNAL DOSIMETRY

Radionuclides may enter the body by ingestion, inhalation, or by absorption through the skin or a break in the skin. This may occur by accident, or by design, as in a nuclear medicine administration.

The question of what happens to the radioactive material, and what radiation doses result to different parts of the body, is a complex one. If important decisions are to be based on the result of an internal dosimetry calculation, expert assistance should be sought.

However, for most routine nuclear medicine administrations, values for doses to various organs have been published for the "average" patient, (actually "Reference Man") and a brief discussion of how these doses are calculated will be instructive.

There are three basic methods for estimating radiation dose in humans:
(1) direct measurements
(2) extrapolation from animal or "phantom" data
(3) calculations based on mathematical models.

Most organs are inaccessible and direct measurements on (or in) them are difficult to perform, so that a combination of methods 1 and 2 must be used for dose estimation. However, the thyroid gland lies close enough to the surface that reasonable measurements can be made on it with a calibrated external probe. As it happens, this is the one organ for which a hospital RSO may be called upon to perform dose estimates, since it is the subject of AECB's Regulatory Document R-58. ***(See chapter B1)*** The calculation of thyroid dose from bioassay results is covered in chapter B6.

Where it is impossible or difficult to determine the activity content of an organ by direct measurement, we must fall back on computer calculations based on models of the physiological behaviour of the radioactive material, hopefully confirmed to some extent by direct measurements in animals, or special dummies, usually referred to as "phantoms".

Early calculation methods began with Marinelli in 1942, and were developed by the originator and many others, notably Quimby and Hine until about 1968. At this time Loevinger and Berman suggested a departure from the old method, which was adopted and developed by members of the Medical Internal Radiation Dose Committee of the Society of Nuclear Medicine, and is hence known as the MIRD method.

For a brief account of the MIRD method, let us start with a simple, specific example. Suppose that we have 10 MBq of Tc-99m uniformly distributed throughout a 2 kg organ, so that all the radiation emitted is absorbed uniformly by the organ. By integrating the decay curve from 0 to ∞ time, we can calculate the total number of disintegrations that occur while the source is in the organ. The equation which describes the decrease in activity with time is $A = A_0 e^{-\lambda t}$, so that the integral mentioned above is $\int_0^\infty A_0 e^{-\lambda t}\, dt = A_0/\lambda = A_0\, T/\ln 2 = 1.443\, A_0 T$, where T is the decay half life, and A_0 is the initial activity.

(See appendix XA1.7 for evaluation of this integral)

The physical half life of Tc-99m is 6 hours. Suppose, also, that the activity is being reduced by excretion from the organ with a biological half life of 10 hours. Then the effective half life will be given by $1/T_E = 1/6 + 1/10$, i.e. $T_E = 3.75$ h (see chapter A1.11).

So, the time integral of the activity, $\tilde{A}$, or "cumulated activity", as it is sometimes called, will be $1.443 \times 10 \times 10^6$ Bq $\times$ 3.75 h $\times$ 3600 s/h $= 1.948 \times 10^{12}$ nuclear transformations.

From the decay scheme of Tc-99m, we can deduce that each nuclear transformation will, on average, release 141.4 keV of energy. So the total radiation energy emitted by all the nuclear transformations will be $1.948 \times 10^{12} \times 141.4$ keV $= 2.81 \times 10^{17}$ eV $= 1.73 \times 10^{-2}$ joules. If this energy is all absorbed uniformly throughout the 2 kg organ, then the absorbed dose is 1.73×10^{-2} joules/2 kg which equals 8.6×10^{-3} grays, or 8.6 mGy. (Recall that by definition 1 Gy = 1 joule/kg).

Unfortunately, the real situation is not quite so simple. In the first place, not all of the radiation energy will be absorbed by the organ containing the Tc-99m - the so called "source organ". Some fraction will escape, perhaps to irradiate a neighbouring organ.

However, if we know the types and energies of the radiations involved, and the dimensions, mass, and

radiation absorption properties of the organ, we can, using a moderately large computer and some sophisticated software, calculate what fraction of the radiation energy will be absorbed by the organ - the so called "absorbed fraction". This has been done by the MIRD organization for the various radionuclides in use in nuclear medicine, and for organs having "average" dimensions and masses. In fact, a whole "average person" has been constructed (on paper) called "Reference Man", and his particulars fed into the computer, so that it can not only calculate the dose to the source organ, but also the doses to surrounding organs ("target organs") from the radiation that escapes the source organ. These absorbed dose values are calculated for unit cumulated activity in the source organ (Bq·s or μCi·h) and published as "S" values.

It is left to the end user in the hospital, to supply the actual cumulated activities in the organs of interest, which will vary widely from case to case, depending on the amount of radionuclide administered, and the biokinetics of the patient or person involved.

A simple example will illustrate the process.

A patient receives an injection of 100 MBq of Tc-99m sulphur-colloid, and we want to estimate the resultant absorbed dose to the liver and the kidneys.

The basic MIRD formulation is $D = \tilde{A} \cdot S$ (target ← source), where D is the mean dose to the target organ, $\tilde{A}$ is the cumulated activity in the source organ, and S (target ← source) is the published S factor for the particular target and source organs in question.

(Actually, the MIRD formulation has changed slightly, but the new one requires more explanation than space permits).

Based on the known physiological behaviour of sulphur colloid, we will assume that 85% of the administered Tc-99m is uniformly deposited in the liver, and that no biological removal occurs, so that the effective half life is equal to the physical half life (6 h). The cumulated activity $\tilde{A}$ is then the time integral of the activity from 0 to infinity. This is 1.443×85 MBq $\times$ 6 h $= 2.65 \times 10^{12}$ Bq·s.

From published tables (MIRD Pamphlet No. 11):

S (liver ← liver) for Tc-99m is
3.45×10^{-15} Gy/Bq·s (4.6×10^{-5} rad/μCi·h),
so the mean dose to the liver is $D = \tilde{A} \cdot S$
$= 2.65 \times 10^{12}$ Bq·s $\times$ 3.45×10^{-15} Gy/Bq·s
$= 9.2 \times 10^{-3}$ Gy (9.2 mGy).

S (kidney ← liver) is 2.93×10^{-6} 1Gy/Bq·s
(3.9×10^{-6} rad/μCi·h), so the mean dose to the kidneys from activity in the liver is
2.65×10^{12} Bq·s $\times$ 3.45×10^{-6} Gy/Bq.s
$= 7.76 \times 10^{-4}$ Gy (0.776 mGy).

The reader is referred to "The MIRD Primer" for a full explanation of the method and further examples of its use.

References

MIRD PRIMER for absorbed dose calculations by Robert Loevinger, Thomas F. Budinger and Evelyn E. Watson in collaboration with the MIRD Committee, The Society of Nuclear Medicine, 136 Madison Avenue, New York, NY 10016-6760.